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FUSION AND FAST BREEDER REACTORS

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PREFACE

This IIASA Research Report compares the liquid-metal fast breeder reactor (LMFBR) and the deuterium-tritium (TOKAMAK) fusion reactor. Both nuclear reactors can potentially produce practically unlimited amounts of energy. This relates exactly to the theme of IIASA's Energy Program, which studies the transition from today's supply systems that use limited but cheap resources to future supply systems that will use unlimited but possibly expensive means for the supply of large amounts of energy.

The conclusions obtained by comparing these two nuclear breeder types are summarized in the first chapter.

It should be pointed out that, in order to delimit the study, it was decided not to write a monograph on breeders in general. For instance, the now often debated concept of hybrid breeders is reported on only briefly. Also, comparisons of costs have been omitted on purpose since they can be made only after a great number of reactors have been sold commercially. This is not even the case with the fission breeder, let alone the fusion breeder whose physics is still under development. The report does include a detailed comparison of the nuclear safeguard features of the two breeder types, but an attempt has not been made to keep up with the development of the last months. Data and developments up to November 1976 that contribute to the comparison of the two breeders have been included.

The draft of this study was distributed to several experts for comments in March 1977. A considerable number of comments have been received, most of which are incorporated in this final version.

Contents

I. INTRODUCTION AND CONCLUSIONS

1. TIME PHASES OF THE ENERGY PROBLEM

It is useful to view the problem of energy supply in terms of three time phases: the near-term phase, a transition phase, and a long-term phase [I-1, I-2].

The dominant issues in the near-term phase, which includes the recent past, involve: *(a)* supply and price of the particular energy sources upon which industrial society has become most heavily dependent, namely petroleum and natural gas; and *(b)* the impact on health and the environment of the immediately available principal supplements to those fuels, namely coal and thermal fission reactors. Included in the issues of supply and price are the political ramifications of the world's growing dependence on a relatively small number of petroleum-exporting nations; the differential impacts of rising energy prices upon rich and poor, as nations and as individuals; and the degree to which a more efficient end-use of energy can be elicited by higher prices and/or regulation without undue economic disruption.

The long-term phase will be characterized by reliance on one or more of the essentially inexhaustible energy sources: solar energy, energy from nuclear fission by breeding, and energy from nuclear fusion (conceivably, geothermal energy from hot dry rock will some day prove to be a viable addition to this group, but that is by no means certain). With the basic constraint of fuel supply removed, the questions that remain will be: *(a)* what mix of these long-term options should be used, and *(b)* at what level should energy use be stabilized? The answers will be governed to some extent by economics and to a very important extent by constraints imposed by the physical and social environments on the handling of very large energy flows with these various technologies [I-3]. These constraints can be viewed as involving the problem of embedding large-scale technological systems into the combination of the atmosphere, hydrosphere, ecosphere, and sociosphere [I-1]. Potentially at risk if the disruptions are too great are the stability of climate, natural services of ecosystems, social structures, and international relations, not to mention direct impacts on health due to leaks of toxic substances from energy systems into the broader environment.

The transition phase is the bridge between the immediate problems of the near-term and the still imperfectly discerned features of the long-term steady state. Of critical importance are the timing of transforming society from reliance on scarce energy sources to inexhaustible ones (including the research strategies needed to bring this about); the extent to which reliance on intermediate sources such as coal, oil shale, and non-breeder fission reactors is justified; the rate of growth of energy use as a whole that is desirable or manageable during the transition period; and the nature of the institutions and mechanisms needed to implement the transition with minimum disruption (including the special problems of decision making in the face of uncertainty). It is important to recognize also that the transition phase involves different timing and different choices in different geographical, cultural, and economic groupings of countries, since both technologies and institutions must be tailored to be appropriate for different physical and socio-economic environments (the latter including goals as well as initial conditions)[I-4].

It is obvious that intelligent energy planning requires accounting for the characteristics of the problem's three time phases, as well as for differences and similarities among various groupings of countries, in an integrated way. The great challenge of the transition period is to devise pathways compatible both with the initial conditions and with the desired endpoints, and having the characteristic that the pathways chosen in different regions are compatible with each other.

2. IIASA AND THE FISSION-FUSION COMPARISON

The International Institute for Applied Systems Analysis (IIASA) was established with the goal of contributing to intelligent planning of responses to precisely the sort of multi-time phase, multi-regional, and multi-disciplinary problems we have just described [I-5]. The Energy Program was a major component of IIASA's efforts from the organization's inception, and the systematic assessment of supply options for the long term was identified early as one worthwhile focus of this project's activities. In this connection, J. Gvishiani, Chairman of the Council of IIASA, observed that it is a confusing and undesirable situation to have vastly different assessments of the fast breeder reactor and the fusion reactor in different groups and countries; the major world-wide challenges ahead of us require that such differences be reconciled insofar as possible, he felt, and, accordingly, IIASA should take the initiative to contribute toward a better and more consistent assessment.

An earlier exchange of views on the fission-fusion comparison, papers by W. Häfele and C. Starr [I-6, I-7], J.P. Holdren [I-8], F. von Hippel [I-9], and Y.V. Sivintsev [I-10], provided a starting point. In subsequent discussion involving primarily W. Häfele, J.P. Holdren, C. Starr, and M.A. Styrikovich it was agreed to undertake a larger study as part of the IIASA Energy Program using, wherever possible, common assumptions and yard-

sticks for the fission and the fusion sides. A team of principle authors was formed that represented, on an equal basis, fission and fusion: W. Häfele, G. Kessler, J.P. Holdren and G.L. Kulcinski. A complementary effort was made in the Soviet Academy of Sciences. As part of the project, two major seminars were held in Moscow (February 1975 and May 1976), providing direction to the study as a whole and an opportunity for coordination and cross-fertilization between the Soviet and Western participants. Besides these two major seminars in Moscow, the team of principal authors met separately at IIASA in Laxenburg, near Vienna, in July 1975 and January 1976, and in October 1976 in Madison, Wisconsin. Some of the Soviet participants spent time at IIASA during August and September of 1976 for coordination with one of the principal authors. In addition to the coordination and interaction made possible by the Soviet participation in the project, some specific written contributions were prepared by Soviet authors; these are included and identified in this report.

It is not the contention of any of the participants that the final report presented here represents a definitive and conclusive treatment of all the issues addressed. Comparisons are difficult by their nature. Even when the technologies being compared have basic similarities in function and approach, as do the two nuclear technologies under consideration here, it is impossible to avoid entirely the situation in which aspects being compared are fundamentally incommensurable--one faces the problem of comparing apples and oranges. A further difficulty is that technical fields related both to fission breeders and to fusion are evolving, sometimes rapidly. This is especially so for fusion because the subject is at so early a stage in its development. We have used the most recent information we could find, but some details without doubt will change even between the final writing and publication. We hope the main conclusions will prove to be more robust, but even there it is possible that some of our conclusions rest on premises which will be shown to be invalid by subsequent work.

Finally, it must be emphasized that, even if our comparison of fusion and fission could be definitive, it would not *by itself* be a suitable basis for deciding whether to develop and deploy specific energy technologies. Answering such strategic questions requires a broader assessment than we have undertaken here, incorporating, among other things, evaluations of energy needs, of other alternative long-term sources such as solar energy, and of transition sources such as coal, oil shale, and non-breeder fission systems.

3. SUMMARY OF THE MAIN CONCLUSIONS

Chapters II through X of the main report address the following topics:

II. The Problem of Fuel Resources
III. Present Status of Fission and Fusion Reactors
IV. Reference Reactor Systems
V. Radioactive Inventories of Reactor Economies
VI. Normal Operating Losses and Exposures
VII. Non-routine Releases
VIII. Safeguards
IX. Materials and Impact of Radiation Damage
X. What Would Be Required for Commercialization? Programs, Timing, and Funding

We present in the following paragraphs a capsule summary of our conclusions on these topics. References and supporting material are found in the corresponding chapters of the main report.

3.1 The Problem of Fuel Resources

Nuclear fuels for fission and for fusion represent energy resources almost incomparably greater than the fossil fuels now relied upon for most of the world's energy use. Fission breeder reactors (breeding plutonium from uranium) and fusion breeder reactors (breeding tritium from lithium) can extract 0.3 to 1.0 MW(th)·day per gram mined of their natural metal fuels, uranium and lithium. Light water reactors extract roughly 100 times less and fossil-fuel burners a few million times less energy per gram of the naturally occurring fuels.

Uncertainty of at least a factor of two to three is evident in estimates by different groups (International Atomic Energy Agency, World Energy Conference, US Energy Research and Development Administration) of the magnitude of US and world uranium resources available at costs of \$66/kg U_3O_8 or less. Uncertainties in the intermediate cost range of \$66/kg to perhaps \$250/kg are even larger. These uncertainties are significant in the context of estimating how long nations could rely on non-breeder reactors to supply a significant part of their energy use, but the uncertainties are *not* significant in the context of the energy potential of fission breeder reactors. This is so because: *(a)* the breeder's high energy extraction per gram of fuel stretches even the smallest estimated quantities of low-cost uranium out to 2000 TW·yr of electricity (1 TW = 10^{12} W = 1,000,000 MW); and *(b)* the insensitivity of electricity cost to fuel cost in the breeder means that high-cost, dilute uranium resources, which exist in quantities far larger than the low-cost resources,

become economically acceptable as breeder fuel (the supply in the oceans alone represents a quantity on the order of 2,000,000 TW·yr of electricity). With fission breeders, therefore, nuclear fuel supply can be considered inexhaustible far beyond any time scale of conceivable planning interest.

The situation is similar for lithium, the limiting fuel resource for D-T fusion. There are uncertainties of a factor of three in US resources available at low cost ($60/kg of Li metal or less), and still larger uncertainties about world resources, but these uncertainties are essentially irrelevant to the outlook for D-T fusion. The *smallest* estimated global quantity of low-cost lithium represents at least 2500 TW·yr of electricity in D-T fusion reactors. Much more expensive lithium could be used economically, including presumably that in sea water, which alone represents at least 100 million TW·yr of electricity. There is no reason to suppose then that lithium resources will limit D-T fusion on any interesting time scale. Deuterium in sea water (by weight) is 200 times more abundant than lithium, and is more economic to extract.

For the fission breeder, the high utilization of ores is reflected in a very low fraction that ore costs contribute to the total busbar cost. These fractions are a few tenths of a percent. This not only makes the low grade ores accessible, but it also provides for a different class of security for ore supply because such low grade ores can be found almost everywhere, and storage of ores does not impose an unacceptable economical burden on the owner of a power plant. Therefore, ore supply embargoes are eliminated and, indeed, the operation of power plants is essentially decoupled from the traditional problems of fuel supply. For the fusion breeder, the situation is fundamentally the same as far as lithium in its function as a fuel is concerned. Indeed, the burn-up ore costs for both breeder types are on the order of 10^{-2} mill/kWh. But, for the case of the fusion breeder, lithium may have the additional function of serving as a coolant, which would raise the total inventory ore costs to a few tenths of a mill/kWh. In any event, for both breeder types, the ore costs are so low that the large existing uncertainties in ore prices that reflect geological and political conditions simply do not matter.

3.2 Present Status of Fission and Fusion Reactors

The development and evolution of new major technologies seem to follow a pattern that distinguishes three thresholds of feasibility:

(a) scientific feasibility;

(b) engineering feasibility;

(c) commercial feasibility.

This pattern is important for judgments or assessments.

The development of fast breeder reactors first followed the line of metallic fuel elements and power-reactor sizes of only few hundred megawatts. It is represented by early concepts such as that of the Enrico Fermi Fast Breeder Reactor (EFFBR). The second line of fast breeder reactor development is distinctly different in terms of fuel technology, reactor physics, safety, as well as power plant characteristics; it uses mixed plutonium/uranium oxide as reactor fuel. After the BN 350 became critical in November 1972, it was the French PHENIX reactor which was the first reactor of the 300 MW(e) class that came to designed power (1974). It has been in full operation on the grid, with load factors in the neighborhood of 85%.* The British PFR and the Soviet BN 350 are also in operation now after initial engineering difficulties, mostly on the steam generator side, have been overcome. The German/Belgian/Dutch fast breeder prototype reactor SNR 300 is in the middle of its construction period; so is the Soviet BN 600, while construction of the US Clinch River Breeder Reactor (CRBR) and the Japanese MONJU reactor is expected to start soon.

The various fast reactor groups of the world now have available a large set of physics and engineering test facilities whose build-up was a major part of the overall effort, in terms of capital investment, manpower and time. The technologies for liquid sodium as a coolant and for mixed oxides as fuels are essentially in hand. A major share of out-of-pile and in-pile tests is devoted to proofing tests as required in the licensing process for large power reactors of the 1200 MW(e) class.

Preparations for the semi-commercial class of 1200 MW(e) are well under way in France, the UK, FRG and the US.

It can, therefore, be concluded that the thresholds of scientific and engineering feasibility of fast breeder reactors have been passed; the threshold of commercial feasibility, however, has not yet been passed. Present projections in the FRG and in France anticipate this threshold for about 1990.

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* PHENIX was shut down at the beginning of October 1976, due to sodium leak in an intermediate heat exchanger which is currently being repaired.

For fusion power, demonstration of scientific feasibility means creating in an experimental device a combination of fuel density, temperature, and confinement time which would lead to a net output of energy in a reactor. No such scientific feasibility demonstration has yet taken place as of late 1976. Of the two main approaches to the problem--magnetic confinement and inertial confinement--magnetic confinement has the longer history (it originated in the early 1950s) and the greater number of variations (TOKAMAK, Mirror machines, and high-density Pinches are the most important at present). The idea of inertial confinement for a fusion reactor dates from the early 1960s, and the two main variations are to use lasers or electron beams to initiate the required implosions. Many proponents of magnetic confinement believe that large TOKAMAK devices, now in the late stages of design or early stages of construction, will achieve scientific break-even (i.e. the Lawson criterion) in the early 1980s (DOUBLET-III and TFTR, for example). Some advocates of laser fusion believe that such systems can also achieve scientific break-even by the early 1980s, although this view is more controversial and is clouded by classification of relevant results. Once scientific feasibility is achieved with either magnetic or inertial confinement, formidable problems of materials and engineering will have to be solved before technological feasibility can be demonstrated in the form of a working reactor. This is unlikely to be achieved before the year 2000. Commercial feasibility will not be assured even when such a reactor exists; it is possible that fusion will work but that it will simply be too expensive. Even if it does prove commercially feasible by the early twenty-first century, a contribution of as much as 10 per cent of the electricity used in industrial nations still seems unlikely before the years 2020 to 2030.

3.3 Reference Reactor Systems

A useful comparison of fusion and fission breeders requires that the analysis be undertaken at a level of detail that can only be provided by reference to specific reactor designs.

The basis for our choice in fission is that the Liquid Metal Fast Breeder Reactor (LMFBR) clearly dominates research and development programs on breeder reactors around the world, making the LMFBR by far the most likely breeder for commercialization. Historically, fast reactors have been preferred to thermal breeders because of their higher breeding ratios, which provide optimum fuel utilization and the possibility of a relatively rapid expansion of the number of reactors. Among fast reactors, liquid-metal-cooled reactors have received much more attention than gas-cooled reactors for partly technical and partly historical reasons. In any case, no prototype gas-cooled fast reactors are under construction at present. Among various existing LMFBR designs we chose the German/Belgian/Dutch fast breeder prototype reactor SNR 300 because we had full access to all the details of the program.

On the fusion side, it is impossible to state with any certainty which configuration will actually lead to a working reactor. At the present time, the TOKAMAK concept seems to provide the greatest promise of success from a scientific standpoint and, therefore, has been the object of most conceptual reactor designs. However, its toroidal geometry and complex magnet configuration make the TOKAMAK a very difficult system to design for electricity production; the construction of such large-scale power plants will undoubtedly be more difficult than that of similar-sized fission breeder reactors. Perhaps some other approach to fusion (Mirrors, laser fusion, etc.) will lead more easily to a reactor than the TOKAMAK concept, but it is too early to say. We have chosen to discuss here the liquid-lithium cooled TOKAMAK because more extensive and detailed information has been accessible for this concept than seems to be available for other approaches.

3.4 Radioactive Inventories of Reactor Economies

It is evident that both fast breeder and D-T fusion reactors will contain high inventories (≈1000 to 5000 Ci/kW(th)) of radioisotopes after a few months of operation. Furthermore, the finite limits on fuel burn-up in fission reactors and finite blanket structure lifetimes in fusion reactors will result in large volumes of high level waste (10 to 100 m^3 per GW(th)·yr). This waste must be properly processed and stored for periods of several thousand years for near-term (stainless steel) fusion reactors, and at least ten times longer for fission reactors. More precise numbers require more precise criteria. The length of surveillance can be shortened to less than 50 years in fusion reactors if certain alloys of vanadium are used. However, the probability of this element being used in early fusion reactors seems small, because very little is known about its properties in a fusion reactor environment, and no commercial industry presently exists to produce the thousand metric ton quantities that would be required for an early fusion economy.

It is also recognized that the inventories of radioactivity in fission and fusion reactors, measured in curies, do *not* provide an adequate basis on which to compare the relative hazards. A somewhat better approach (although still imperfect because it does not include the pathways to release--see Chapter VII) is to use the Biological Hazard Potential (BHP). The BHP index incorporates the effects of radioisotopes on humans either by inhalation from the air or by ingestion through the water route. The inhalation BHP is mainly applicable in the event of an accidental release of radioisotopes either from the reactor or during any of the reprocessing steps before they are inserted into the final waste storage location. The ingestion BHP is mainly applicable to the long term storage of wastes but could also be important in specific accidental releases. On the basis of *inventory* alone (disregarding the probability of release, which is treated in chapters VI and VII), we find that the BHP for inhalation is one to two orders of magnitude higher for the

LMFBR system than for fusion up to the point of the reprocessing of fuel wastes. After most (99%) of the Pu isotopes have been separated from the spent fuel, we find that the BHP for ingestion of the *inventory* (not the amount that would necessarily be released) is a factor of two to 100 times higher for fission than fusion for the first 1000 years after shut-down. For the next million years, the BHP for ingestion of the stored wastes is two orders of magnitude higher for fission than the structural material of a stainless-steel fusion reactor.

The economic incentive to reprocess fission fuels soon after discharge in large reprocessing facilities, serving about 10 to 30 GW(e) FBR plant capacity, means that large amounts of high level wastes must be handled, transported, and eventually solidified for long-term storage. Fusion systems have an integrated fuel cycle (tritium separation); and after appropriate compaction, structural steel and other waste material can directly be stored as solids. This tends to reduce the potential for a release of radioisotopes to the environment and could lessen the hazard potential associated with the final transportation of fusion reactor wastes to the ultimate storage facilities.

Finally, there is the question of measuring the total burden of radioisotopes to society. If one considers the BHP integrated over the lifetime of the various isotopes, allowing for the fact that any given isotope could pass through several biological generations, one finds that stainless steel D-T fusion reactors represent a factor of ten smaller burden than an LMFBR on the basis of a unit of energy produced. The use of vanadium alloys could increase this advantage for fusion to two orders of magnitude. As mentioned above, the probability that such alloys can indeed be used in economic fusion power reactors is much smaller than for similar reactors with a steel structure. However, hopes remain that structural materials for fusion with even better activation properties than vanadium, such as perhaps titanium alloys, will eventually be shown to be feasible.

3.5 Normal Operating Losses and Exposures

Because the inventory of tritium is likely to be around 250 MCi/GW(e) in fusion compared to around 0.025 MCi/GW(e) in the LMFBR, the degree of tritium control would have to be about four orders of magnitude tighter in fusion to meet the same requirement on environmental doses from tritium (achieving 5 mrem/yr at the fencepost would mean tritium control to about 1 part in 10^4 per year for a 1 GW(e) fusion plant).

For the case of the fission breeder, the limiting factor is the release of α-emitters, iodine 129, and krypton. In all cases the reprocessing facility appears to make the largest contribution. Confinement factors (annual flow/annual release)

of $2 \cdot 10^9$ for transuranium emitters, $2 \cdot 10^4$ for iodine, and 10 for krypton would permit meeting the regulations now under consideration in the US. As long as confinement factors of 10^{10} can be achieved, the fuel fabrication facilities do not seem to contribute significantly to the overall releases. The fast reactor itself is a small contributor compared to both reprocessing plants and fuel fabrication plants. All these confinement factors appear to be within reach.

Comparing fusion and fission with respect to routine releases, the degree of control required in the most sensitive part of the fuel cycle (tritium in fusion reactors, transuranium α-emitters and iodine 129 in fission-fuel reprocessing plants) appears to be attainable, but it still must be demonstrated in daily operation of large facilities. It also remains to be seen in both cases what the cost burden associated with these controls will be.

It is important to recognize the magnitude of the impact on technology of regulations concerning releases from the fast-breeder fuel cycle. It is necessary to define clearly the nature of such regulations and the specific levels that will have to be met, so that the technologists can adjust their designs accordingly. This illustrates a more general observation. While originally the inherent technical characteristics of nuclear power shaped the development of the technology, it is now more and more also the nature of regulations and standards which is shaping its development.

3.6 Non-routine Releases

Early concerns about the safety of LMFBR focused on control characteristics and the possibility of core recompaction in accidents that begin with sodium boiling and local fuel melting. These concerns were accentuated by the emphasis on compact cores and metallic fuel elements in breeder designs of the 1950s and early 1960s. For the case of the large cores and mixed-oxide fuels typical of all prototype and commercial LMFBR designs in the 1970s, it is now known that in their crucial respects the control characteristics are substantially similar to those of the LWR. Moreover, a large and growing body of theoretical and experimental evidence supports the view that the propagation of local fuel failures in a way that leads to recompaction in the large-core, mixed-oxide fueled LMFBR would require combinations of events and degrees of spatial and temporal coherence that are not physically realistic.

The large LMFBR prototypes that are in operation in France and in advanced stages of construction in the FRG have undergone licensing reviews as stringent with respect to safety as the ones that are applied to the LWR. The design basis accidents (DBA) for these large LMFBR encompass the possibility of failure of both independent shut-down systems, following a hypothetical large insertion of reactivity or coast-down of the main sodium pumps.

The calculated consequences of melting and core disassembly in these maximum hypothetical accidents define the design characteristics of the containment systems required for licensing (strength of reactor vessel and primary piping; strength and leak rates of surrounding double steel and concrete containment structures). In addition to the pressure loads during a DBA, large commercial LMFBR would also have to cope after a DBA with long-term cooling of large masses of molten and dispersed fuel. While this capability appears to be at hand for the 300 MW(e) class LMFBR, additional development work is needed for larger LMFBR power stations. Meeting these design requirements, which as the French and German experience indicates can be done with reasonable technical effort, can restrict radiation doses to 1 rem or less at the plant boundary in the event the DBA occurs. The overall conclusion is that the LMFBR can meet the same predetermined safety standards as are applied to other fission reactors. This will also hold for the LMFBR fuel cycle (fabrication and reprocessing plants).

In the case of fusion, reactor safety analysis is necessarily much more primitive because the technology cannot yet be described in detail. Examination of stored energies and potential pathways for energy release in conceptual Controlled Thermonuclear Reactor (CTR) designs indicates that sudden failures of the magnet support and vacuum systems could produce enough mechanical energy to severely damage the reactor. Loss of coolant or coolant flow coupled with failure to shut down the fusion reactor could cause local interior structural damage. The characteristics of fusion plasmas and the very small amount of fuel present in the reaction chamber at any time mean that reactivity accidents will not be an important concern. Decay-heat due to neutron activation of structural materials is small enough in most designs to be substantially easier to handle than in fission reactors. For CTR designs where liquid lithium serves as breeding medium and coolant, the very large chemical energy stored in this coolant and the high flame temperature of the lithium-air and lithium-water reactions (somewhat worse in both respects than the sodium in a comparable LMFBR) probably represent fusion's most important vulnerability to accidents capable of releasing sizable quantities of radioactivity. Both LMFBRs and liquid-lithium cooled CTRs require careful design of steam generators to handle safely the possibility of leaks that bring water into contact with liquid metal.

Many of the possible accident pathways for CTR can be minimized in importance by intelligent design, which includes the apparent possibility of tritium breeding in ceramic lithium compounds, and cooling with pressurized helium instead of liquid lithium. Such an approach may also be able to reduce the inventory of blanket tritium that could be released in an accident, but other sources of tritium in proximity to the blanket (i.e. vacuum pumps, diverter collector plates, etc.) will not be affected by the change to solid breeders. Enthusiasm about the potential flexibility in design of CTR must be tempered with the recognition that there may be important trade-offs--for example, the probable need to use toxic and relatively scarce beryllium for neutron multiplication if solid breeders are employed in realistic blanket designs. Designers of fusion

systems can anticipate that the approaches they devise to control energy release from magnets, vacuum systems, coolant, and so on will doubtless be subjected to much the same critical scrutiny and demand for high reliability experienced now in fission-reactor licensing proceedings.

It is possible to make a very crude comparison of fission and fusion reactors with respect to the consequences of events worse than the design basis accidents--resulting, for example, from acts of war, sabotage, or hypothetical events exceeding the design capabilities of the safety systems. Applying the consequence model of the Reactor Safety Study (Rasmussen Report) of the US Nuclear Regulatory Commission shows that hypothetical release of a substantial fraction of the fission products and 0.5 percent of the actinides in an LWR (release PWR-1 in the Reactor Safety Study) would produce roughly 100 times more early deaths under adverse meteorological conditions than a release of 10^8 Ci of tritium oxide from a CTR under the same conditions. Much in need of further investigation is what fraction of the activation products in a CTR and of the actinides in an LMFBR could be released in such hypothetical events, as these could significantly affect the calculated outcomes. Comparative examination of delayed as opposed to early casualties is also needed.

In the CTR most of the fuel cycle is within the reactor containment structure in the form of the tritium cycling systems. For the LMFBR there must exist in addition fuel reprocessing and fuel fabrication plants, and the potential for accidents on the way to or at these facilities needs careful examination. Such analysis is not yet nearly as refined as that of the LMFBR itself. Both LMFBR and CTR will require some form of radioactive waste management facilities, for which accident analysis will also have to be done.

3.7 Safeguards

Understanding and comparing safeguards aspects of fission and fusion can be facilitated by making several important distinctions:

- spread of knowledge vs. spread of nuclear material;
- explosive vs. radiological threats;
- vulnerability of power reactors vs. that of the rest of the fuel cycle;
- diversion (countered by detection) vs. theft (countered by protection);
- offenses by governments vs. offenses by private groups;
- national vs. international controls.

As fission power spreads, the associated spread of bomb-related material is more important than the spread of bomb-related knowledge. (This is a problem of all forms of fission, not just breeders.) If fusion power spreads, the associated spread of bomb-related knowledge (in the inertial confinement approach) would be more important than the spread of bomb-related material. However, neutrons from any D-T fusion reactor could be used by the operators to produce fissile material; this raises the question of "safeguarding neutrons".

Misuse of nuclear material as a radiological rather than an explosive weapon is a threat associated with both fission and fusion. (Here fusion's hazard is from the tritium associated with all approaches, not just the inertial confinement approach.) With respect to airborne dispersal of plutonium or tritium, fusion appears to have a quantitative advantage over fission of two to five orders of magnitude, depending on the chemical form of the materials; with respect to waterborne dispersal, the two are approximately even.

In the case of fission power, safeguarding other steps in the fuel cycle is a greater problem than safeguarding power stations. Moreover, the fission fuel cycle is spread out (requiring transportation of nuclear materials), and it will remain so unless nuclear energy centers become the norm. In the case of fusion, the tritium for the most part remains in the power stations; transportation is necessary only when new power stations are being started up.

The detection of diversion of nuclear material is of concern between governments, and the Non-Proliferation Treaty is addressed to this issue; it calls more for international than national controls. The protection of nuclear material from theft is a problem for individual nations and calls more for national than international controls.

For a state, the most direct route to the fabrication of a few crude nuclear explosive devices is probably the construction of centrifuges; it is not the deployment of economically significant civilian nuclear fission power. Without adequate safeguards, the latter may, however, be a permanent temptation to divert some nuclear material, possibly at a later date.

Technical, managerial and institutional measures can increase the efficiency of safeguards. Establishing an equilibrium of any kind between risks, benefits and costs is, in the first analysis, a step that entails social and political considerations as well as technical insights. The question has been raised whether adequate safeguards would entail too heavy social costs, for instance infringements of civil liberties. We have not felt capable of properly addressing this issue here.

3.8 Materials and Impact of Radiation Damage

It is clear that one of the major factors limiting the efficiency and economic viability of both fission and fusion reactors is the degradation of materials performance in the reactor environment. The reduced component lifetime affects the economies of nuclear power plants in six major areas:

(a) reduced thermal efficiency (lower operating temperature);

(b) reduced plant factors (to change damaged components);

(c) increased capital costs (for remote handling equipment);

(d) increased operating costs (for component replacement and manpower);

(e) increased volume of radioactive wastes;

(f) increased demand for scarce elements.

These problems have been studied for over 20 years for the LMFBR and resulted in the choice of PuO_2/UO_2 as a fuel, SS 316 as a cladding and core structural material, and B_4C as a control rod material. However, it is quite probable that even these materials will not be sufficient for a completely economical breeder economy, and carbide fuels and high nickel-base alloys are being investigated for possible long-term application.

The process of selecting the optimum structural materials for fusion reactors is, by comparison, in its infancy. Early reactor designs almost exclusively used Nb alloy, but present thinking strongly suggests the use of austenitic steels at least for the first generation of power reactors. However, due to the high helium production rates, the pulsed nature of most viable reactor concepts, and the extreme reliability that will be demanded on the reactor components with respect to vacuum leaks and dimensional stability, it is now widely accepted that most of the reactor components will not last the lifetime of the power plant. The necessity to *quickly* replace damaged components in a very high radiation environment will put a severe strain on the design of a fusion power plant.

Both fission and fusion structural components and fuels share some of the same intrinsic radiation-damage problems. Void swelling in metals is probably more important in LMFBR because of the close tolerances for coolant flow, but high-temperature helium embrittlement will definitely be a greater problem for D-T fusion reactors than for fission because of the higher energy neutron spectrum. Irradiation creep will prove to be a major problem in both types of reactors because of: *(a)* high displacement rates in the LMFBR, and *(b)* the high thermal stresses in a fusion reactor. Fatigue is likely to be more severe in fusion reactors, especially in inertially confined systems.

Suggestions have been made to use alloys other than the austenitic steels in fusion reactors, mainly for the purpose of reducing the radiation levels in the wastes. However, essentially all of the proposed alloys suffer from one or more serious deficiencies in the fusion reactor environment. Aluminum alloys are subject to even more modest temperatures. The refractory metals Nb and V offer higher temperature operation with less irradiation-induced embrittlement, but they are extremely susceptible to pick-up of interstitial impurity atoms, which also causes embrittlement, and they suffer from a lack of a commercial industry to supply a mature fusion economy. Molybdenum alloys are probably the best suited of the refractory metals but require major advances in joining of large-scale reactor components. For these and other reasons, it is of debatable validity to compare fusion-reactor systems based on these untested and ill-understood materials to workable LMFBR reactors. For the foreseeable future, fusion reactors based on austenitic steels will probably be the standard.

Because of the low power density in the blankets of all fusion reactors (not just the TOKAMAK), the nuclear-island requirement for materials is likely to range from 10 to 40 t/MW(e) of steel versus 3 t/MW(e) in fission breeder reactors. In addition, the replacement of damaged structural components may range from 0.1 to 0.5 t/MW(e)·yr of steel for fusion compared to 0.06 t/MW(e)·yr for fission breeder reactors. These large material resource requirements for sometimes rather scarce materials will certainly be a greater problem in a fusion economy than in a fission economy. Careful attention will have to be paid to methods of reducing those requirements for fusion, or they could prove to be the limiting factor to the amount of energy that can be produced by fusion, despite essentially unlimited fuel resources. For example, one material which has been proposed for fusion reactors utilizing solid breeder blankets is beryllium. If this element is required, the ultimate amount of fusion generating capacity that can be built could be limited to as little as a few TW(e), and the electricity generated to a few hundred TW(e)·yr. (Total world generating capacity in 1976 is around 1.5 TW(e), and annual use around 0.8 TW(e)·yr.)

We conclude that materials problems are much more diverse and severe in fusion reactors than in fission reactors. Without intensive long-range development programs it is possible that fusion may never transcend the engineering feasibility phase into a commercial regime.

3.9 What Would Be Required for Commercialization? Programs, Timing, and Funding

It is expected that it will take 50 to 60 years to pass through the scientific and engineering feasibility stages before demonstrating commercial feasibility of the fast breeder reactor starting from 1942. This is based on the belief that commercial feasibility of fast breeder power stations could be attained between 1990 and the

year 2000, while the related fuel cycle services are expected to require an additional 10 to 15 years.

Approximately the same time frame is anticipated for fusion reactors (although the uncertainty is greater): commercial feasibility of large-scale power stations could be achieved in the time period of 2010 to 2020. Unlike the LMFBR, there are no fuel-cycle-related services which will be required after a large number of fusion reactors are constructed, and, therefore, the point of commercial feasibility should be easier to define.

Three generations of reactors seem to be required for both types of energy sources to demonstrate commercial feasibility:

- experimental power reactors (10 to a few 100 MW(th));
- prototype or demonstration reactors (250 to 500 MW(e));
- semi-commercial reactors (1000 to 1500 MW(e)).

Along with these major facilities, there are a large number of smaller but equally important test facilities that need to be developed for:

- physics;
- engineering;
- materials;
- safety.

Beyond the physics facilities, the materials testing facilities can be particularly costly and time consuming to the overall program development. There is hardly any way to circumvent these problems, as each generation of reactors requires higher performance characteristics which are difficult to test in facilities existing up to that point. In order to be useful, fast breeders and fusion reactors must fit into existing schemes and rules of electricity production. Demonstration of availability, maintainability, and repairability is in itself a complex procedure that requires time. Especially important is the aspect of licensing. In fact, the rules and fundamental data underlying the licensing process must be developed almost in parallel with the reactors and facilities that are so to be licensed. Aspects of public acceptance broaden the scope even further.

Finally, the fuel cycle that serves fast breeder reactors must also be developed. By necessity, the hot part of the fuel cycle can be developed on a technically significant scale only when irradiated fuel is available in significant quantities from reactors whose acceptability often seems to require the services of such fuel cycle. In fact, the LWR seems to be presently in such a situation. The problems of final waste disposal extend the time horizon even further.

In the US, with a broad and stretched-out development program, more than ten billion dollars are expected to be necessary for reaching commercial maturity of LMFBR. By contrast, in European countries, the development programs seem to be less broad and less stretched out and, thereby, seem likely to be considerably cheaper. The difference in funding points to the degree of flexibility that such programs seem to have. In any event, it must be borne in mind that there are parallel development programs in the world whose positive interactions contribute significantly to each other. The value of such positive interactions is rather high. In fact, that may explain the lower cost that European programs seem to have when compared with the US program.

The situation for the fusion program is much less well defined, but recent projections in the US program reveal that it may require 20 to 25 billion dollars to bring fusion through the demonstration power reactor phase, and it is not unreasonable to expect that another five to ten billion dollars will be required to progress through the commercialization stage. In contrast to the breeder program, the European fusion program is much smaller and of a longer time duration. The Soviet program is approximately the same as the US program in level of effort now, and it is expected to keep pace with the US program. Therefore, it is reasonable to expect that--worldwide--it may require as much as 50 billion dollars to reach commercial feasibility of fusion. The same benefits for international cooperation in fusion research as in fission are expected to allow for considerable flexibility in design and should increase the probability of long-term success.

4. OVERVIEW

The foregoing section summarized, on a topic-by-topic basis, the specific conclusions of our comparison of LMFBR and D-T fusion. We attempt here to integrate these results and insights into an overview of the significance of these two technologies and their role in the evolving time phases of the energy problem.

Of central importance is that both fission breeders and D-T fusion have the potential, in terms of fuel supply, of providing very large amounts of electricity almost indefinitely. In this respect, there is no difference between them that is of any practical significance. To answer the question of fuel supply for the indefinite future is an enormous benefit, but the benefit has its price. Both for fission breeders and for fusion, the price includes: a heavy investment in research and development before the commercial stage is reached; continuing high capital costs for the commercial reactors and supporting facilities; and a commitment to maintain a high degree of meticulousness and vigilance in the construction and operation of these technologies and the sequestering of their wastes.

Notwithstanding significant differences in the basic physical processes of fission and fusion, the presently envisaged technologies of using these processes for electricity production have much in common: complex large-scale engineering based on large, central-station power plants; material damage and activation by neutrons; the need to contain inventories of radioactivity within the plant and to manage radioactive wastes beyond the lifetime of the reactors; and, for many present designs, use of liquid-metal cooling and heat-transfer technologies. At the same time, the nature of the fusion process in principle allows for a degree of flexibility in the technologies used to harness it. This flexibility, if explored and utilized, offers the possibility for D-T fusion to be quantitatively superior to fission in important environmental respects, despite the qualitative similarities already mentioned.

Specifically, D-T fusion has the potential for quantitative advantages in the form of: lower hazard potential in its radioactive inventory (and, accordingly, smaller predicted consequences of hypothetical large releases); lower radioactive decay-heat; smaller hazard potential and shorter hazard lifetime associated with radioactive wastes; less shipment of dangerous material outside the reactors; and smaller hazard potential for use of tritium as a radiological weapon (compared with plutonium in fission). There are, of course, qualitative differences in accident pathways in fusion and fission. With respect to the spread of the capability of making nuclear bombs, we conclude that fission spreads relevant material more than knowledge and fusion spreads knowledge (related to the inertial confinement approach) more than material.

It must be emphasized that achieving the potential environmental advantages of fusion in a practical system will require that high priority and prolonged attention be given to environmental characteristics from the earliest stages of designing fusion systems. The advantages will not materialize automatically simply because fusion is fusion. It is possible to envision fusion systems in which many of the most important environmental advantages compared to fission do *not* materialize.

Those environmental advantages of fusion that *are* achieved will have to be weighed against the cost of achieving them. No such weighing can be done today, both because the technology has so far to go (the thresholds of scientific and engineering feasibility have yet to be passed), and because the value that society will place on such advantages has yet to be determined.

The LMFBR, by contrast, has passed the thresholds of scientific and engineering feasibility, with commercial feasibility still to be demonstrated. Herein lies a dilemma of timing. The LMFBR meets the fundamental requirement for long-term energy sources--namely, a nearly inexhaustible fuel supply--but the timing of its development has been such that the LMFBR's commercial feasibility, as well as its environmental and social

characteristics, are being judged against the yardstick of existing energy technologies of only short-term or transitional significance such as oil, natural gas, light-water reactors, and coal. Such comparisons are relevant for helping to determine the appropriate timing for commercial introduction of a technology such as the breeder, but since oil, gas, LWR, and coal do not meet the basic fuel-supply criterion as long-term sources, they do *not* provide suitable yardsticks for judging the LMFBR's viability and desirability as a way to meet these long-term needs.

Just as part of the present predicament of the LMFBR arises from evaluating a long-term source against short-term competitors, so also is there a related pitfall (another confusion of time perspectives) that could damage the future of fusion. The pitfall is that the desire to bring fusion to commercial fruition in time to compete in the transition time frame (say, in the period 2000 to 2030) may lead in fusion programs around the world to a disproportionate emphasis on early engineering feasibility at the expense of potential environmental advantages. If fusion technology is steered too early in the direction of doing whatever seems necessary to produce commercial power, the field may be shaped for a long time to come by approaches that exclude the principal environmental benefits that represent fusion's greatest asset as a long term energy source. This would be a serious misfortune.

It is essential, therefore, to try to keep separate in technology assessments the differing requirements of the short-term, transitional, and long-term phases of the energy problem. The most significant comparison of long-term sources is with each other, as we have done here with LMFBR and D-T fusion. As the needed information becomes available, such comparisons should be extended to include large-scale use of solar energy and perhaps fusion and fission fuel cycles other than D-T and plutonium-burning LMFBR.

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The foregoing conclusions have been written and subscribed to by all four principal authors, W. Häfele, J.P. Holdren, G. Kessler and G.L. Kulcinski. While the main body of the report that follows reflects a degree of collaboration throughout, it should be obvious that principal responsibility for authorship of different sections was divided according to the interests and backgrounds of the authors--W. Häfele and G. Kessler accepting the primary role on the fission side, and J.P. Holdren and G.L. Kulcinski on the fusion side. The responsibility for the individual contributions of the Soviet authors, which are interspersed in the text, is, of course, that of the authors of these contributions. The principal authors wish to acknowledge their gratitude to Ch. Starr, Palo Alto, Calif., USA, whose continued interest and support is appreciated.

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All opinions expressed herein are those of the authors as individuals and should not be construed as representing positions of any of the organizations with which the authors are associated.

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II. THE PROBLEM OF FUEL RESOURCES

W. Häfele, J.F. Holdren, G.L. Kulcinski

1. BREEDING REACTIONS

1.1 Fission Breeder Reactions

In the case of fission reactors, two kinds of reactions lead to breeding. The chain for the production of plutonium from uranium 238 is:

$$U238 + n \rightarrow U239$$

$$U239 \xrightarrow[\beta-]{23.5\ min} Np239$$

$$Np239 \xrightarrow[\beta-]{2.33\ day} Pu239$$

U238 is not fissionable by neutrons below 1.4 MeV, while Pu239 is fissionable by all neutrons. By capturing one neutron the fertile U238 is transformed into the fissionable Pu239. If this neutron originates, for example, from the fission of this Pu239 nucleus, it may be possible to combine the chain reaction and the breeding cycle [II-1].

Similarly, in the case of thorium, the breeding reaction is the following:

$$Th232 + n \rightarrow Th233$$

$$Th233 \xrightarrow[\beta-]{23.5\ min} Pa233$$

$$Pa233 \xrightarrow[\beta-]{27.4\ day} U233$$

Again, if this neutron originates from the fission of a U233 nucleus, the chain reaction and the breeding cycle can be combined [II-2].

Under real technological reactor conditions, plutonium is always a mixture of Pu239, Pu240, Pu241, and Pu242. It should be realized that under such circumstances Pu240 must be considered a fertile material. Pu240 cannot be fissioned by thermal neutrons. Neutron capture, however, leads to the fissile Pu241 isotope:

$$Pu240 + n \rightarrow Pu241$$

Pu241 is unstable and undergoes a β-decay with a half-life of 13.2 years. This is long enough for it not to impair the value of this breeding reaction.

1.2 Fusion Reactions

The reaction almost certain to serve as the principal energy source in first-generation fusion reactors is that of deuterium with tritium:

$$D + T \rightarrow n(14.1\ MeV) + He4(3.5\ MeV)$$

The energy released amounts to 94,000 thermal kilowatt-hours [kWh(th)] per gram of reacting nuclei; this is roughly 1800 times the energy that must be supplied to the reactants in order to initiate fusion. Because tritium is almost nonexistent in nature (global natural inventory of approximately 70 kg), it must be bred. The main breeding reactions involve the neutron-induced fission of lithium:

$$Li6 + n \rightarrow T + He4 + 4.8\ MeV$$

$$Li7 + n \rightarrow T + He4 + n - 2.5\ MeV$$

The overall reactions for D-T fusion are therefore effectively:

$$D + Li6 \rightarrow He4 + He4 + 22.4\ MeV\ (75{,}000\ kWh(th)/g\ reactants)$$

$$D + Li7 \rightarrow He4 + He4 + n + 15.1\ MeV\ (45{,}000\ kWh(th)/g\ reactants)$$

The He4 "ashes" of the D-T fuel cycle are nonradioactive, as are both isotopes of lithium. Radiological considerations enter, however, in the form of tritium (its half-life is 12.3 years) and the copious production of 14 MeV neutrons (this is a direct radiation hazard during reactor operation and the source of induced radioactivity in the reactor structure).

Additional fusion reactions of deuterium with itself and with He3 will play a secondary role in first-generation fusion reactors, and perhaps a primary role in more advanced systems. These reactions are:

$$D + D \rightarrow He3\ (0.82\ MeV) + n\ (2.45\ MeV)\quad (22{,}000\ kWh(th)/g\ reactants)$$

$$D + D \rightarrow T\ (1.01\ MeV) + H\ (3.02\ MeV)\quad (27{,}000\ kWh(th)/g\ reactants)$$

$$D + He3 \rightarrow H\ (14.67\ MeV) + He4\ (3.67\ MeV)\quad (98{,}000\ kWh(th)/g\ reactants)$$

Accordingly, burning deuterium to completion would give:

$$6D \rightarrow 2He4 + 2H + 2n + 43.2\ MeV\quad (96{,}000\ kWh(th)/g\ D)$$

The two D-D reactions occur with roughly equal probability. The peak cross-sections for these and the D-He3 reaction are considerably smaller than those for D-T, and the reactant energies corresponding to the cross-section maxima are much higher than for D-T. Achieving a net energy gain from the D-D reaction chain therefore will be significantly more difficult than for D-T; the ratio of the energy of the reaction products to the energy supplied to the reactants in the D-D chain is about 70, a margin about 25 times smaller than for D-T.

Plotted in Figure II-1 are the fusion reaction-rate parameter $\langle\sigma v\rangle$, and the figure of merit $\langle\sigma v\rangle E_f/E_o$ for the principal fusion reactions. Here σ is the fusion cross-section for particles moving with relative velocity v; $\langle\ \rangle$ denotes an average over Maxwellian velocity distribution functions for the reacting species; E_o (which is identical to kT for a Maxwellian velocity distribution) is the mean kinetic energy of the reactants; and E_f is the energy yield per fusion reaction.

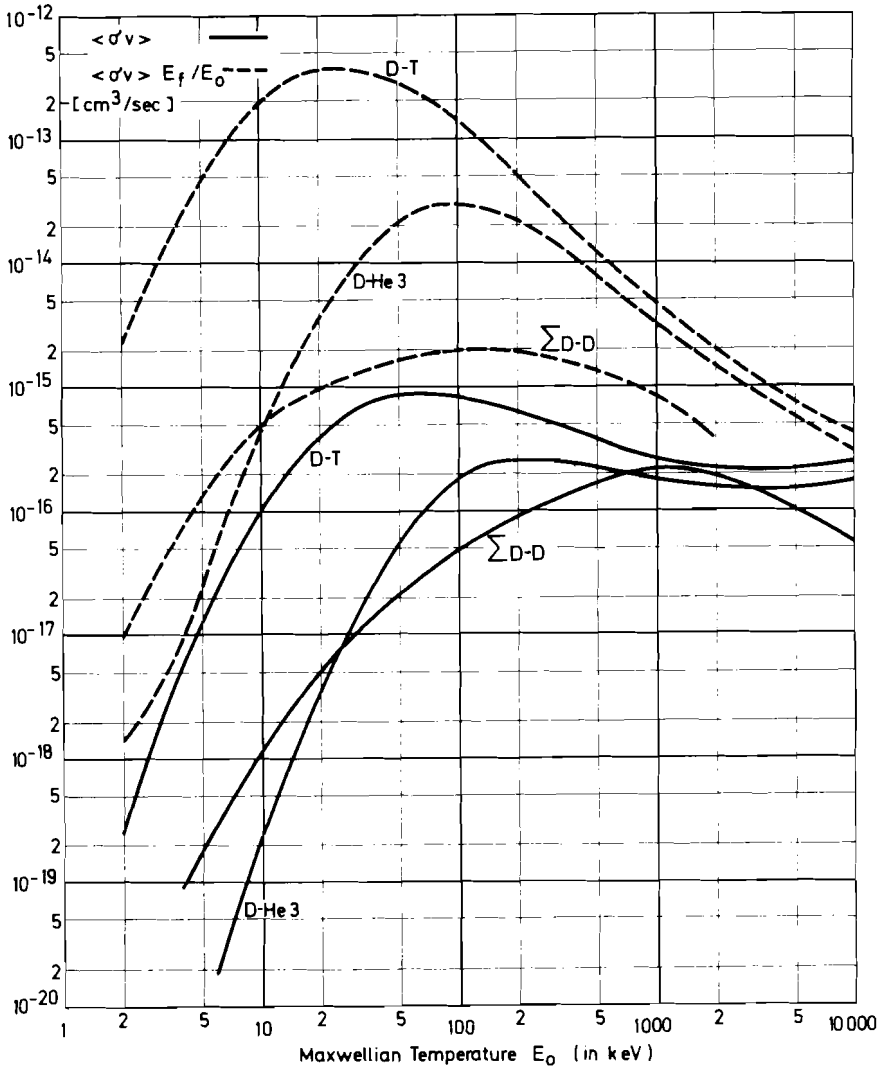
Other more exotic fusion reactions exist: for example, H-Li6, H-Li7, and H-B11. These reactions have higher Coulomb barriers than the others discussed here. They therefore require higher mean reactant energies for ignition, and their energy gains are smaller. These reactions are potentially attractive in environmental terms because they produce no neutrons. But we do not consider them in detail here because they appear so difficult to harness, and because little detailed information about them is available.

2. A REVIEW OF RESOURCES

2.1 Uranium and Thorium Resources

Reviewing uranium resources is a constant and open-ended process. More recently, there have been three studies:

(1) The joint OECD/IAEA report in its latest version of December 1975. It takes into account only uranium resources of the price category below \$ 30/lb U_3O_8 [II-3].



Fusion yields (E_f) are: D-T = 17.6 MeV, D-D = 3.65 MeV, D-He3 = 18.3 MeV.

Note: D-He3 reaction cannot realistically be considered separately from D-D, because no independent source of He3 is known, and because it must be produced by D-D reactions.

Figure II-1: Reaction-Rate Parameter and Energy Production Figure of Merit for Principal Fusion Reactions

(2) The *Survey of Energy Resources 1974* by the Ninth World Energy Conference, Detroit, [II-4].

(3) The *Report of the Liquid Metal Fast Breeder Reactor Program Review Group*, (ERDA-1 report of January 1975 [II-5]. This report incorporates results of the Preliminary National Uranium Resource Evaluation Program (PNURE), and uses a classification of resources of its own.

For the present purposes it suffices to report on the results of these studies in aggregate form.

The OECD/IAEA study gives values not only for the US but also many other countries. One can obtain a total figure for the world with the exclusion of the Soviet block and China. The data published by the World Energy Conference are essentially consistent with those of the OECD/IAEA study. The ERDA-1 report employs much more recent data but is limited to the United States. By comparing the US data in this report with those of the OECD/IAEA study, one can get an idea of (non-communist) world figures. Similarly, such comparison also gives a clue as how to extrapolate from the \$ 0-30/lb U_3O_8 range to uranium of much higher cost classes. Tables II-I and II-II are made for such comparisons and extrapolations.

Besides uranium there is thorium. The World Energy Conference of 1974 also identified thorium reserves. These are given in Table II-III, which uses the same format as Tables II-I and II-II, and which also gives the uranium data for the purpose of comparison. The thorium data are by far more uncertain because

Table II-I: Uranium Resources[a] (OECD/IAEA, [II-3]) (1000 t U)

	Reasonably Assured	Estimated Additional	Total
USA	454	812	1266
"World"[b]	1810	1680	3490

[a] Price range: \$ 0-30/lb U_3O_8 in 1975 dollars;

[b] World being the total of the following countries: Algeria, Argentina, Australia, Brazil, Canada, Central African Republic, Denmark, Finland, France, Gabon, Federal Republic of Germany, India, Italy, Japan, Korea, Mexico, Niger, Portugal, South Africa, Spain, Sweden, Turkey, United Kingdom, United States, Yugoslavia, and Zaire.

Table II-II: US Uranium Resources (ERDA, [II-5])
(1000 t U)

Cost Range[a] ($/lb U_3O_8)	Identified	Potential	Total
0-15	400	1200	1600
15-30	140	930	1070
at 100	Chattanooga shales 60-80 ppm		4000
at 150	Chattanooga shales 25-60 ppm		6000

[a] Costs given here are the so-called forward costs, which do not include operating costs and capital investments incurred prior to the time the resource estimate is made. Thus they are lower than total extraction costs and probably much lower than the price at which the uranium would be sold.

Table II-III: Thorium and Uranium Resources (World Energy Conference of 1974, [II-4])
(1000 t U)

		Identified	Potential	Total
USA[a]	Thorium[c]	320	?	-
	Uranium	730	1300	2030
World[b]	Thorium[d]	800	1960	2760
	Uranium	1960	2080	4040

[a] Data from Bureau of Mines, Bulletin 650, 1971;

[b] Total of 28 non-communist countries in the case of U, and a total of 14 non-communist countries and the USSR in the case of Th, including the US;

[c] Only up to $ 10/lb;

[d] Cost range mostly unknown.

much exploration is still going on. The same is true for low-grade uranium; data for the Chattanooga shale are only representative. The fission breeder does have the potential of using such low-grade uranium. Its use in LWRs recently has also been studied by J.P. Holdren [II-6]. The World Power Conference of 1974 observes [II-7]:

> *"At costs up to $ 200 per kilogram of uranium, the amounts of uranium available are in tens to hundreds of megatonnes and at costs to $ 500 per kilogram in thousands of megatonnes. In the present period, during which only non-breeder reactors are commercially available and for which relatively low-cost nuclear fuel is required, much further exploration and development of resources will be necessary. Later when breeder reactors become predominant, economic uranium (and thorium) resources will become near-infinite in extent."*

For reasons of completeness it must be mentioned that there is uranium in seawater (but no thorium.) The average concentration is at 3.4 µg/liter. Using a seawater volume of $1.4 \cdot 10^{18}$ m^3 this gives almost $5 \cdot 10^9$ t uranium. The related technology is still highly uncertain, however. This is reflected in the uncertain costs of such uranium. Estimates range from $ 35 to $ 300/lb U_3O_8. At the recent IIASA Conference on Energy Resources, A. Brin reported on these uranium resources [II-8]. His conclusion is as follows:

> *"The importance of the reserve built by the uranium in solution in the oceans, should not blind one to realities. According to one of the classifications mentioned by M. Grenon, it can be said that:*
> *(1) This reserve is proved;*
> *(2) It is 'submarginal'."*

2.2 Resources of Deuterium and Lithium

One in 6700 atoms of hydrogen in seawater is deuterium. This amounts to 33 grams of D per m^3, or a total resource of $4.6 \cdot 10^{19}$ g in the oceans of the world. Burning this amount of deuterium to completion in the D-D reaction chain would produce $4.4 \cdot 10^{24}$ kWh of thermal energy, an amount equivalent to some $60 \cdot 10^9$ times the world energy use in 1975. At the market price of deuterium prevailing in the early 1970s, i.e. $ 0.30 to 0.40 per gram, the contribution of the raw fuel to the cost of electricity in a D-D fusion reactor would be on the order of 0.01 mill/kWh(e).

For the D-T fuel cycle--in which the raw fuels are deuterium and lithium--it is the lithium supply that limits ultimate energy production. The amount of energy that will be obtained in practice from each gram of natural lithium (7.4% Li6, 92.6% Li7) depends on the details of the geometry and composition of the breeding region in each reactor, including the chemical form of

lithium, the degree of enrichment in Li6 (if any), the presence of neutron-absorbing and neutron-multiplying materials, and the method of utilizing the excess tritium from the breeder reactors. It is shown in Appendix II-A that there is no unique value for Li but a whole range exists depending on parameters as those listed above. However, the most reasonable value based on the present concept of utilizing liquid lithium to both cool the reactor and breed the tritium reveals an energy content of 12,000 kWh(th)/g of natural lithium. This assumes a total breeding ratio of 1.3 after all neutron leakage, non-productive breeding zones, and toroidal neutronics are considered. Such an energy content may be too high as it presumes that all tritium can be instantaneously transferred between reactors (no delay), that all the tritium atoms that are bred can be burned (not used to fulfill the inventory requirements of new plants) and that the excess tritium is fed into plants utilizing depleted Li (which is pure Li7 essentially) for cooling and some further breeding of tritium (referred to hereafter as a "near-breeder"). If one were to simply use the excess tritium in "burner reactors" where no more tritium is bred, then the energy content would drop approximately to 9500 kWh(th)/g of natural Li. Finally, if one were to use solid breeders containing Li highly enriched in Li6, then the energy content would approximately drop to 5700 kWh(th)/g for a breeding ratio of 1.15 and a breeder/burner economy. If a "breeder/near-breeder" combination were used with solid breeders, the energy content could be as high as 6000 kWh(th)/g of natural Li.

Lithium resources have been reviewed by a number of authors in the past several years [II-9 to II-14]. Most attention has been given to the United States, where exploration for lithium has been somewhat more extensive than elsewhere. Specialists appear to agree, however, that exploration for lithium has not been exhaustive in the US nor elsewhere, because the existing reserves are very large compared to the conceivable non-nuclear requirements for many decades to come. In this situation, there is little incentive to conduct expensive exploration programs to find and classify additional reserves. Should the success of D-T based controlled fusion eventually generate very high demands for lithium, it is generally agreed that the resulting further exploration would yield large additional deposits.

The present status of lithium supplies in various categories, as appraised by various reviewers, is summarized in Table II-IV. Substantial uncertainty has been introduced into the picture for US reserves at present prices by a recent report [II-13] that, at these prices, less than a tenth the amount of Li previously estimated will actually be extractable from the Silver Peak Nevada brines. From the standpoint of fusion, however, the lithium price could increase two to three times with little effect on the price of electricity. At the recent market price of $ 0.02 per gram of Li metal, burn-up of Li at 12,000 kWh(th) per gram in a fusion reactor producing 0.4 kWh(e)/kWh(th) would correspond to 0.004 mills per kWh(e), excluding carrying charges on lithium inventory. The highest Li inventory requirement of

Table II-IV: Lithium Resources

Nature of Deposit	ppm Li by Weight	10^{12} Grams of Contained Li, Range of Estimates	References
Silver Peak (Nevada) brines (present prices)	300	0.04-0.49	[II-12, II-13]
All US brines (to-3-times-present-prices)	35-300	3.9-4.8	[II-9 to II-14]
US pegmatites (to-3-times-present-prices)	6000-7000	0.94-1.2	[II-9 to II-14]
US total, (to-3-times-present-prices)	-	4.8-6.0	[II-9 to II-14]
Canadian and African pegmatites (to 3-times-present-prices)	5500-22,000	0.34	[II-13]
Chilean brines (present prices)	2000?	1.1?	[II-13]
Non-US world total (to-3-times-present-prices, exc. USSR, China)	-	1.4-2.0	[II-9, II-13]
World oceans (price uncertain)	0.17	240,000	[II-9]

any fusion-reactor design published to date is about 1.2 t/MW(e) [II-13], which contributes 23 US dollars/kW(e) at $ 0.02 per gram.* Clearly, a tripling in Li prices could easily be tolerated, so it makes sense in the fusion context to look at more than the reserves at present prices. In Table II-IV, the category labeled "to-three-times-present-prices" is synonymous with the classification "known plus inferred or conditional reserves", used by many reviewers. The fraction of those reserves that

* The lowest inventory is approximately 0.2 t/MW(e) (see Chapter IX).

will actually be recoverable by a given date--say, the year 2000--depends on details of technology and economics that cannot be accurately predicted [II-14].

US Li reserves to-three-times-present-prices would support the inventory for 2.7 to 5 million MW(e) of fusion generating capacity at 1200 kg/MW(e), or a burn-up corresponding to 15 to 30 quadrillion (10^{15}) kWh(e) at 12,000 kWh(th) [4800 kWh(e)] per gram. The generating capacity stated is 7 to 13 times the total US electricity-generating capacity in 1975, or about three times that of the world in the same year. The intermediate generation figure of $20 \cdot 10^{15}$ kWh(e) corresponds to roughly 10,000 times the electrical energy generated in the US in 1975.

The consumption of lithium in the US in non-nuclear applications stood at about $2.4 \cdot 10^{6}$ kg/yr in 1970 [II-9], corresponding to 0.004 of the US reserves at present prices (taking the lowest estimate for the Silver Peak brines) and $\leq$ 0.0005 of the US reserves at to-three-times-present-prices. Lithium consumption for these non-nuclear uses has been increasing at eight to ten per cent per year, however, and applications in catalysis and advanced batteries suggest that this growth may continue for some time [II-15]. Nevertheless, the present high ratio of reserves to consumption, and the good prospects for further discoveries upon additional exploration make it unlikely that an absolute shortage of lithium could constrain the prospects of D-T fusion in any time frame of practical interest. (Naturally, a total failure to carry out additional exploration and mine development could lead at some point in the future to temporary Li shortages, but such short-sightedness should be avoidable.)

The resource picture in the very long term (perhaps beyond a hundred years from now) is somewhat obscured by two uncertainties. The first is whether the D-D fusion fuel cycle becomes successful; if it does, the need for lithium is eliminated and the enormous and already economically accessible supply of D in seawater removes all fuel supply constraints for millions to billions of years. If D-D fusion does not work, the situation in the very long term depends on whether Li can be extracted economically from seawater. At 0.17 ppm by weight, Li in the ocean is about 200 times less concentrated than D, but some 50 times more concentrated than uranium. Extraction of uranium from seawater is already being seriously considered, and very tentative cost figures mentioned fall in the range of $ 80 to $ 670 per kilogram U [II-8]. Should Li be extractable at $ 100 to $ 200 per kilogram, which is certainly not inconceivable, the associated cost of burn-up in fusion reactors would still be negligible. The contribution of this expensive Li as inventory at 1200 kg/MW(e) would be in the range of $ 100 to $ 200 per kW(e), providing some incentive to work out designs with smaller inventories (e.g., using Li only for breeding T but not as the coolant). This is even more important in solid-breeder designs when the initial lithium breeder compounds may not last for the lifetime of the reactor. (See Chapter IX).

3. BUS BAR SENSITIVITY TO FUEL COSTS

3.1 Fission Breeder Reactors

Fusion breeders and fission breeders make use of isotopes that are abundant in nature; thus a situation arises where the supply of primary fuels is no longer a determining parameter. In the following the fuel cycle costs of breeders are discussed.

In the case of fusion breeders, attention has been given not only to the ore costs for first inventories and of consumption but also to the costs of processing the fuel throughout its cycle. This has been done because the development of fusion breeder reactors is still in the early phases, when indicators as to the costs involved are necessary information. The fission breeder is a different matter; there are not only the aspects of conceiving a fuel cycle but also an operational and technical experience at different levels. Such experience is significant in the case where the LWR fuel cycle experience can be applied, for the LWR *and* FBR fuel cycles are basically the same. For this very reason we are aware of the differences between costs and prices, between a concept and its commercial reality.

So it makes sense to give a break-down of electricity production costs at bus bar as shown in Table II-V [II-16].

Table II-V: Electricity Production Costs for Various Fuels (US mill/kWh(e) for new plants, at 7000 h/yr load factor)

	Lignite	Hard Coal	Oil/Gas	Nuclear Power (LWR)
Fuel Including Fuel Cycle	6.8	23.2	19.2	5.5 to 7.2
Operation and Maintenance	2.0	2.8	1.2	2.4
Capital Cost	9.6	6.4	6.4	14.4
Total	18.4	32.4	26.8	22.3 to 24.0

Source: Data published by German utility RWE, January 1976.

It refers to the situation in the FRG as of February 1976. It is striking to see the high fraction of fuel costs for hard coal and oil/gas. In the case of the FRG, there are finite resources of 10^9 t of lignite that allow for a fraction of fuel costs that is significantly lower and compares with the low fraction of LWR fuel cycle costs. Such LWR fuel cycle costs are made up by ore costs, enrichment and processing costs, including final waste disposal. The value of 7.2 mill/kWh(e) in Table II-V already reflects the recent sharp increases in cost for ore, enrichment, and processing.

The fuel cycle costs of 5.5 mill/kWh(e) for LWRs of Figure II-2 relate to present realistic costs for:

LWR Fuel Cycle Cost Parameters	
natural uranium	30 \$/lb U_3O_8
Pu costs	27 \$/g Pu
conversion to UF_6	4 \$/kg U
enrichment (tails assay 0.25%)	100 \$/kg SWU
fuel element fabrication	120 \$/kg fuel
transport, reprocessing, waste disposal	360 \$/kg fuel
price escalation 5%,	
interest rate 9%.	

Figure II-2 also compares fuel cycle costs of fast breeders. It explains that there are essentially three almost equal contributions to the total fuel cycle costs of fast breeders:

- fuel element fabrication;
- interest costs for the plutonium inventory;
- transport, reprocessing, and waste disposal costs.

The cost break-down of Figure II-2 relates to cost parameters of the LMFBR fuel cycle which appear attainable under future commercial conditions, all other cost parameters being equal to LWR parameters:

LMFBR Fuel Cycle Cost Parameters	
fuel element fabrication	1200 \$/kg core fuel
transport, reprocessing, waste disposal	1100 \$/kg core fuel
breeding ratio	1.17

The assumed breeding ratio of 1.17 is rather pessimistic. Values up to 1.35 for the LMFBR appear attainable. With such

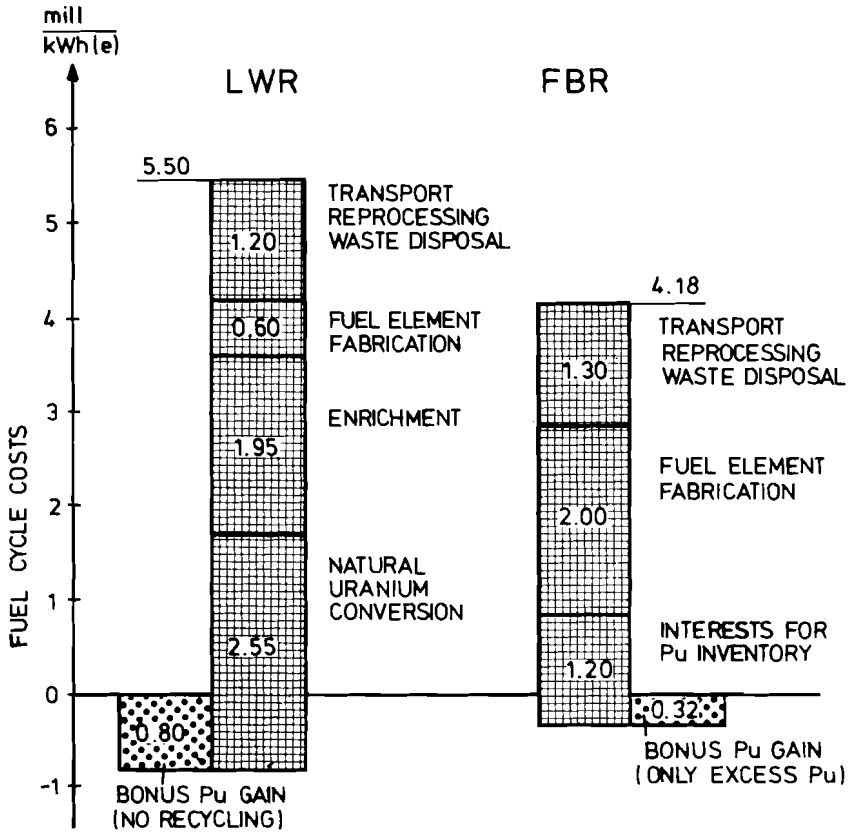


Figure II-2: Structure of Fuel Cycle Costs for LWR and LMFBR

higher ratios the Pu gain would increase and the total fuel cycle costs decrease.

It is not the purpose of this paper to go into the technical and commercial details of fuel cycle costs. Instead we refer to a survey publication [II-17]. The point is rather that in the case of the fast breeder, ore requirements will go down by a factor of 60 to 100, and enrichment services will be eliminated. Therefore, a total of only 70 per cent of LWR fuel costs may, pessimistically, be expected. This would be equivalent to roughly 4 to 5 mill/kWh(e).

From one gram of natural uranium one can extract $0.95/(1 + \alpha)$ MW(th)·day, if α is the ratio between capture and fission cross-sections. Assuming a value of 0.15 for α one arrives at an equivalent of roughly 20 MWh(th), which is higher by 70 per cent than the figure for fusion, which is approximately

12 MWh(th) per gram of natural Li mined. Let us continue to assume that $ 80/kg of natural uranium is a high but not unreasonable price for uranium in the foreseeable future. Let us further assume that 80 per cent of this amount can be fissioned in a fast breeder reactor. Then 1 kg of U mined relates to $0.8 \cdot 10^3 \cdot 20$ MWh(th), or $ $1.6 \cdot 10^5$ (at 25 mill/kWh(e)). This must be related to the cost of 1 kg of natural uranium mined of $ 80. The ratio is 2000.

Besides the yearly requirements for burn-up, there are inventory requirements. A 1000 MW(e) fast breeder reactor may have an inventory of 50 t of natural uranium. At $ 80/kg U this relates to $0.6 \cdot 10^6$ $/yr at 15 per cent annuity. This compares to an income of $1.75 \cdot 10^8$ $/yr (25 mill/kWh(e) and 7000 h/yr). This ratio is 300. Uranium inventory costs are, therefore, significantly higher than burn-up costs, but both costs are indeed *negligible*.

It is a deep-seated feature of breeders that they are essentially decoupled from the supply of fuel resources. The results of these considerations are summarized in Table II-VI.

Table II-VI: Fast Breeder Fuel Cycle Costs

Total Fuel Cycle Costs	mill/kWh(e)	≈5
Burn-up Ore Costs	mill/kWh(e)	$1.5 \cdot 10^{-2}$
Burn-up Ore Costs Relative to 25 mill/kWh(e)		$0.6 \cdot 10^{-3}$
Inventory Ore Costs[a]	mill/kWh(e)	$0.9 \cdot 10^{-1}$
Inventory Ore Costs Relative to 25 mill/kWh(e)[a]		$0.36 \cdot 10^{-2}$

at $ 80/kg natural uranium

[a] at 7000 h/yr load factor and 15% annuity.

3.2 Fusion Reactors

It is commonly assumed that the fuel cycle costs of a D-T fusion reactor are negligible (usually less than 0.01 mill/kWh(e) [II-18]. Such an assumption is true if the costs are only based on the amount of fuel consumed and if factors such as enrichment, fabrication, inventory, and reprocessing costs are not included. We have attempted to examine this subject in more detail than have previous reports, and to also examine the sensitivity of the electrical generation costs to changes in fuel prices. This analysis requires a great number of assumptions which we will carefully list so that as the picture becomes clearer one may update these numbers.

The first step in this type of analysis is a listing of the burn-up rates of the various fuels* and an inventory of "fuel" materials. Such a compilation is given in Table II-VII. We have chosen to use two types of fusion reactors as examples; the liquid-lithium-breeding scheme (UWMAK-I [II-19]) and the solid-breeding scheme (UWMAK-II [II-20]). The burn-up numbers are consistent with the reactor operating parameters listed below:

	Liquid-Breeder (UWMAK-I)	Solid-Breeder (UWMAK-II)
maximum breeding ratio	1.49	1.16
energy per neutron, MeV	20.08	21.56
kWh(e) per year	$1.03 \cdot 10^{10}$	$1.2 \cdot 10^{10}$

The burn-up numbers for Be include the following reactions: (n,2n), (n,γ), (n,p), (n,T), and (n,α).

The materials inventory is quoted in terms of MW(th) and again represents the natural elemental amounts, because enrichment factors tend to cloud the picture. For example, most solid-breeders for fusion reactors require that the lithium be enriched in Li6, and values of approximately 90 per cent enrichment are typical. This means that one gram containing 90% of Li6 requires approximately 12 grams of natural Li (7.42% Li6) to be mined. We have also assumed that the

---

* Here, fuels in fusion are interpreted loosely in order to include Be, which generally is only present in a fusion reactor when there is a need to achieve breeding ratios greater than 1, with solid-lithium-breeding materials.

Table II-VII: Fuel Requirements for Current Nuclear Reactor Designs

	Burn-up mg/kWh(e)[a]	Inventory kg/kW(th)
Fusion--Liquid[b]		
Lithium	0.53	0.23[e]
Deuterium	0.013	$1.2 \cdot 10^{-5}$
Fusion--Solid[c]		
Lithium	0.55	0.11[f]
Deuterium	0.010	$1.9 \cdot 10^{-5}$
Beryllium	0.03	0.08
Fission (LMFBR)[d]		
Uranium	0.125	
Plutonium		$1.5 \cdot 10^{-3}$

[a] Natural isotopic abundance;

[b] UWMAK-I [II-19];

[c] UWMAK-II [II-20];

[d] French SUPERPHENIX class;

[e] Natural Li equivalent;

[f] Natural Li equivalent.

solid-breeder and Be multipliers have lifetimes as long as the reactor, and that no replacement is required. This assumption will be examined later. Finally, the deuterium inventory was calculated on the basis that there would be one day's throughput in the fuel cycle, and that one would require a ten day supply of D_2 in reserve. This means that for our reference UWMAK-I reactor the D_2 inventory is 61.6 kg, and it is 94.3 kg for UWMAK-II; the difference in inventories comes mainly from the burn-up fraction per pass.

It is interesting to note that the amount of Li required to be mined from out of the ground for liquid systems is only twice that required for solid-breeders. This is due to a complex balance between increased parasitic absorption, neutron production by the Be, softened spectrum, and essentially no tritium from Li7 in the solid-breeder case. If the solid-breeder does not last for the plant lifetime (on account of burn-up, swelling, sintering, etc.), then the total plant demand on Li resources could actually be larger in the solid-breeder case, even if one chose to reprocess the breeder material.

The next step in this analysis is to calculate the contribution of fuel burn-up alone to the overall electricity production costs. The following raw material costs are used:

	$/kg
Lithium	20
Deuterium	440
Beryllium	220
(90% Li6) $LiAlO_3$	35

The calculations (see Table II-VIII) show that approximately one half to two thirds of the fuel costs for burn-up lie with lithium, and only one third with the burn-up of deuterium. The burn-up of Be contributes approximately 30 per cent to the cost of fuel in the solid-breeder system.

A more realistic assessment of fuel cycle costs should include the inventory costs of the breeding material; i.e., liquid lithium or the solid-breeder/berylllium combination. It is straightforward to calculate the appropriate numbers for the solid breeder and neutron multiplier if one can assume a full reactor lifetime. After that time it is assumed that the "fuel" components can be stored until the activity that is due to contamination and impurities decays away. We will use a 15 per cent per year carrying charge for this calculation, which includes payment of the original material, enrichment, and fabrication costs as well as a reasonable return on capital. Both the direct and indirect charges should be considered, and we will use 35 per cent for the indirect costs, although others might use higher numbers.

The situation for liquid lithium is not so clear because it serves a dual role, being breeder and coolant. For example, the thickness of the Li zone in the UWMAK-I blanket is 55 cm (1159 t), and there is another 541 t in the piping and heat exchangers. For the purposes of these simple calculations we will attribute the Li inventory costs only to that lithium in the blanket, and the Li external to that will be charged to the reactor itself. The inclusion of this inventory cost for Li contributes another 0.46 mills per kWh(e) to the fuel cycle costs, while the inventory cost for D_2 only contributes another 0.0005 mills per kWh(e) to the burn-up costs of 0.0057 mill/kWh(e).

The inventory costs for solid-breeder fusion reactors are about four times as high as for liquid systems mainly because of the cost of the neutron multiplier beryllium. This amounts to almost three-fourth of the fuel cycle costs and could increase if the low level of Be reserves are depleted by a large-scale fusion economy [II-21].

Table II-VIII: Summary of Fuel Cycle Costs (mill/kWh(e)[a])

	Burn-up Only	First Core Loading[b]	"Total" Fuel Cycle for Finite Component Lifetime[c]
Fusion: Liquid Lithium			
Lithium	0.011	0.47	0.47
Deuterium	0.0057	0.0062	0.0062
Tritium extraction equipment	-	-	0.14
Total	0.017	0.48	0.62
Fusion: Solid Breeder			
Lithium (enriched $LiAlO_3$)	0.011	0.36	1.43[d]
Deuterium	0.0044	0.0051	0.0051
Beryllium	0.0066	1.62	4.77[e]
Tritium extraction equipment	-	-	0.19
Total	0.022	1.99	6.4

[a] Using raw material costs stated in the text;

[b] Includes direct and indirect (35%) raw material costs, fabrication, enrichment, burn-up, and a 15% carrying charge over the life of the plant;

[c] Includes all in-first-core loading, reprocessing or replacing of the solid-breeder and neutron multipliers every two years;

[d] Assume that costs are depreciated over the core lifetime (2 years);

[e] Same assumption as [d], but the reprocessed and refabricated price is $ 150/kg including 4% loss during refabrication.

The final way of estimating fuel cycle costs would be to include *(a)* fuel extraction equipment (either from the breeder or the coolant system), and *(b)* any reprocessing costs associated with a reduced lifetime of the fuel components.

We have chosen to use a 35 per cent indirect cost factor for these studies, although more conservative estimators may argue for 50 per cent or higher. The results can be scaled accordingly if the reader wishes to use a new number.

The fuel extraction costs involve items such as absorbers, regenerators, distillation columns, diffusion windows, fuel injectors, reprocessing of divertor-particle collection streams, etc. The estimates for these components can only be order of magnitude numbers at this time because of the uncertainty of the final reactor configuration, and because so many components in a fusion reactor play a dual role. For example, how much of the vacuum system should be charged to the fuel cycle, or how much of the divertor-particle collection system should be included? Both systems are necessary for a TOKAMAK to operate and yet must be a part of the overall fuel system. We have tried to extract appropriate numbers and realistic fractions of costs attributable to fusion from the UWMAK-I and UWMAK-II cost studies [II-22], and have come up with the following approximate numbers for the fuel-extraction cycle only:

	Direct Capital Costs
UWMAK-I	$ 7 million
UWMAK-II	$ 11 million

The need for reprocessing of the breeder can cause a considerable increase in the fuel cycle costs of a fusion reactor. This does not apply to a liquid Li system, because the burn-up rate is so small that it hardly affects the neutronics of the plant over a 30 year lifetime. However, solid ceramic breeders can undergo a great deal of swelling due to helium gas generation, sintering, and restructuring over periods as short as a year or two [II-23]. The same situation can occur in Be [II-24], so that it is probably safe to say that these two components of a fusion reactor blanket will have to be replaced after a year or two of operation (we have arbitrarily chosen two years in this calculation), which means one needs 14 "cores" in addition to the first loading.

The next question to ask is whether one can afford to simply dispose of the old "fuel" and replace it with fresh fuel, or whether one would wish to reprocess the radioactive material, adjust the enrichment, and refabricate the elements. This is especially critical for beryllium because of the limited resource picture. Considering how expensive remote handling facilities are, we have assumed that one would simply replace the $LiAlO_3$ with new material each time and one would reprocess the Be. Estimates of reprocessing costs are as high as

$ 150/kg (versus $ 220/kg of new material) because of the desire to use powder metallurgy techniques to form a porous solid. We have also assumed that there would be a four per cent loss during each reprocessing operation.

The "total" fuel cycle costs are now increased by 30 per cent for liquid systems and by a factor of approximately three for solid-breeders. Obviously, the use of solid-breeders can significantly increase the fuel cycle costs and must be carefully considered, before it can be used to offset higher tritium blanket inventories.

A summary of the fuel cycle costs, obtained by using the above reasoning, is given in Table II-VIII. The first column represents the traditional method of estimating fuel costs, and the figures are approximately 0.02 mill/kWh(e). When one includes the fuel inventory, the fuel cycle costs increase by a factor of approximately 30 for liquid systems and a factor of 90 for solid systems. Finally, inclusion of the tritium-processing equipment and the replacement of cores raises the fuel cycle costs to 0.62 mill/kWh(e) for the liquid system and to over 6 mill/kWh(e) for solid-breeders. This latter value represents a 300-fold increase over the traditional method of quoting fuel costs.

The final area to consider is the sensitivity of the numbers in Table II-VIII to fuel prices. We have chosen to quote this sensitivity in terms of the mill/kWh(e) increase per doubling of current fuel price values. The results are given in Table II-IX.

The burn-up costs in both types of systems and the first-core loading costs in the liquid-lithium system are directly proportional to the cost of the fuel. The first-core loading and the "total" fuel cycle costs for the solid-breeder include the enrichment costs, which presumably would be independent of the raw materials cost. Hence the increase in the cost of electricity is approximately 60 per cent of the fuel cost increase. Finally, one contribution of Be to the total costs is through the 4 per cent loss factor that is assumed during the reprocessing and refabrication stage. This only contributes approximately 0.24 mill/kWh(e), and the rest of the increase (0.07) is due to the increased cost of two initial cores--one for loading and the other to be used when the first is reprocessed--spread out over the reactor lifetime.

The important conclusion to be drawn from Table II-IX is that the doubling of fuel costs in liquid lithium systems contributes less than 0.5 mill/kWh(e) to the cost of electricity, and this figure is dominated by the cost of Li.

The second point is that solid breeders are more sensitive to fuel prices by a factor of two or more, depending on the lifetime of the breeder and neutron multiplier.

Table II-IX: Effect of Doubling Fuel Prices on the Cost of Electrical Power Generation in Fusion Reactors

Increase in Costs over those in Table II-VIII (mill/kWh(e))

System	Burn-up Only	First Core Loading[a]	Total Fuel Cycle Costs[a]
Fusion: Liquid			
Lithium	0.011	0.47	0.47
Deuterium	0.0057	0.0062	0.006
Total	0.017	0.48	0.48
Fusion: Solid			
Lithium	0.011	0.21	0.86[b]
Deuterium	0.0044	0.0051	0.0051
Beryllium	0.0066	0.74	0.31[b] [c]
Total	0.022	0.96	1.18

[a] Amortized over lifetime of plant except where noted;

[b] Average value over 30 years, cost of core repaid in two years from purchase;

[c] Including 4% loss during reprocessing.

The calculations of a two-year lifetime show that a 1.2 mill/kWh(e) increase in electricity costs might be expected per a doubling of fuel prices for solid-breeder systems.

While this analysis clearly favors liquid over solid-breeder systems from an economic argument, it shows that the resource demand is about equal for both systems, and the solid-breeders would be favored if tritium inventory in the blanket alone was a consideration (remember, less than ten per cent of the T inventory is normally in the blanket of a magnetically-confined fusion reactor). Nevertheless, one can now say that the contributions of the fuel cycle to the costs of liquid-metal, magnetically-confined fusion reactors are still small but not negligible. The fuel cycle costs of a solid-breeder system will be higher and could be as much as five per cent of the total costs of generating electricity.

4. TOTAL NUCLEAR FUEL RESOURCES AND ORE REQUIREMENTS

The amounts of energy available in various nuclear and non-nuclear energy sources are tabulated for comparison in Table II-X, wherein present and hypothetical future energy consumption rates are also given.

Table II-X: Some Fuel Resources and Consumption Rates (in 10^{12} GJ = 10^{21} J = 0.948 • 10^{18} BTU = 0.95 Q) [II-25, II-26]

	Energy Content
Fuel Resources	
World uranium, to $ 33/kg, used in LWR	2
World oil, ultimately recoverable, 1976	13
World gas, oil, coal, ultimately recoverable[a]	80
US lithium , at BR = 1.25	250
US uranium, to $ 250/kg, used in LMFBR	700
World lithium, to $ 60/kg, BR = 1.0[b]	7,600
World uranium, to $ 250/kg, used in LMFBR[b]	9,000
Uranium in oceans, used in LMFBR	200,000
Uranium in continental crust, to 500 m, used in LMFBR	10,000,000
Lithium in oceans, 12,000 kWh(th)/g	11,000,000
Deuterium in oceans	16,000,000,000
Consumption Rates	
World energy use, 1975	0.26
US energy used for electricity, 1975	0.02
Total annual energy use, 8 • 10^9 people, at 6 kW/cap[c]	1.5

[a] Reference [II-26] emphasizes the distinction between recoverable and geological resources;

[b] Extrapolated from US on the basis of land area, see [II-27];

[c] Rate corresponds to Sweden, 1973.

In addition to amounts of contained energy, of course, it is also of interest to know how much material (ore) must be handled to make the energy available. Ore requirements for the various energy sources, normalized to the amount that would be required for each source, if it alone had to provide the 1975 world electrical energy generation (about 600,000 MW(e)•yr), are given in Table II-XI.

Table II-XI: Ore Requirements to Produce 1975 World Electrical Energy by Various Processes

(600,000 MW(e)•yr per year at assumed efficiency of 40% = 1,500,000 MW(th)•yr per year)

Process	Ore (Fuel ppm by Weight)	Ore Required (10^6 t)
D-D Fusion	Seawater (32 ppm D)	4.2
D-T (Li) Fusion	Pegmatite (6000 ppm Li)	0.18
	Silver peak brine (300 ppm Li)	3.7
	Seawater (0.17 ppm Li)	6,400
LMFBR Fission	Colorado sandstone (2000 ppm U)	0.51
	Chattanooga shale (60 ppm U)	16.5
	Seawater (0.003 ppm U)	330,000
LWR Fission	Colorado sandstone (2000 ppm U)	44
	Chattanooga shale (60 ppm U)	1,400
	Seawater (0.003 ppm U)	28,000,000
HTGR Fission	Chattanooga shale (60 ppm U)	700
Coal Fired Steam	Bituminous coal	1,700

Assumed energy contents for these tables are:

	Assumed Specific Energy Content kWh(th)/g
Terrestrial and oceanic lithium Breeding ratio 1.3	12,000
Uranium, LMFBR Fissions 60% of natural U	13,200
Uranium, HTGR Fissions 1.4% of natural U	308
Uranium, LWR Fissions 0.7% of natural U	154
Coal	7.75

5. ANOTHER APPROACH TO THE ASSESSMENT OF URANIUM RESOURCES AND FUTURE URANIUM-ORE REQUIREMENTS

(A.M. Belostotsky)

All the calculations for the whole world mentioned above are based on the national calculations of individual countries. The assessments given by these calculations are not satisfactory because of the following points:

(1) They do not take the whole world into account;

(2) The calculations for different countries are made with different degrees of reliability;

(3) A more or less careful assessment has been made only for resources that are commercially efficient.

But during the energy crisis it became evident that the upper limits of what are considered commercially efficient (feasible) resources can be raised rapidly. Aspects of the energy crisis included the movement of nuclear fuel prices toward their maximum permissible levels, which also had an effect on coal-fired plants. Coal is cited here because coal resources are the only conventional resources which can provide for long-term energy development and thus--if corresponding environmental-protection measures are taken into account--can be treated as a possible substitute for nuclear energy. Another important aspect of the crisis was the marked stimulation of new discoveries of nuclear fuels. But for now it is too early to sum up both these aspects. Thus in order to assess uranium resources, we will use the latest data given by national sources and international organizations (IAEA), and the methodology described by A. Alexandrov and N. Ponomarev-Stepnoy [II-27, II-28].

For an assessment of the uranium resources in the world, the authors of the section above took the USA as a base, a country with a rather large, geologically-varied territory in which a rather good job of investigating uranium resources has been done.

Let us assume for the USA that reasonably assured resources of U_3O_8 up to $ 30/lb are equal to $454 \cdot 10^3$ t of U, and that estimated additional resources within the same cost range equal $812 \cdot 10^3$ t [II-3]. This gives a total of $1266 \cdot 10^3$ t of uranium.

Since the ratio between the growth of reserves and the cost of new discoveries has not tended to decrease in previous years, it is possible to consider these figures as the lowest estimation. Next, generalizing these figures to all the earth's territory excluding Antarctica and the sea shelf, one gets a total of world uranium resources equaling $17.5 \cdot 10^6$ t. Surely there can be no final agreement as to whether this figure is an upper or lower estimation, but there are some good reasons to consider it a lower estimation:

(1) With regard to the world the uranium resource figure, the US basis is taken at random and has no peculiarities. Rather it seems that uranium resources are rather uniformly spread throughout the world;

(2) We took only those US resources which have already been discovered and, as mentioned, there are good prospects for further discoveries;

(3) There is a good chance that the upper cost limit for available uranium resources will probably rise, because the limit for conventional resources is also rising.

Thus we feel that a value of $17.5 \cdot 10^6$ t of U at present-day costs is not excessively optimistic, though it is five times higher than that given by the IAEA in its latest *Report on Uranium Resources* [II-3].

Taking into account the assessment of world uranium resources of $17.5 \cdot 10^6$ t, it is possible to compare this figure with the demand.

There are many projections of future uranium needs. For our calculations we have used the latest data issued by the IAEA [II-29]. This organization predicts that the world's total nuclear power plant capacity will be at the levels shown in Table II-XII.

Let us take the capacity of 2000 GW(e) predicted for the end of this century and assume that 200 t of natural uranium is required per year for 1 GW(e) in thermal reactors. This gives a consumption level of 400,000 t of uranium per year in

Table II-XII: Forecast of Total World-wide Nuclear Power Plant Capacity.

Year	1975	1985	2000
Nuclear Power GW(e)	70	500	2000

the year 2000. In the case of uranium resources totaling $17.5 \cdot 10^6$ t, it is clear that such a nuclear option involving uranium ore would be available only for several decades. Thus the development of nuclear energy based solely on thermal reactors will not lead to unlimited energy resources.

6. CONCLUSIONS

Nuclear fuels for fission and for fusion represent energy resources almost incomparably greater than the fossil fuels now relied upon for most of the world's energy use. Fission breeder reactors (breeding plutonium from uranium) and fusion breeder reactors (breeding tritium from lithium) can extract 0.3 to 1.0 MW(th)·day per gram mined of their natural metal fuels, uranium and lithium. Light water reactors extract roughly 100 times less and fossil-fuel burners a few million times less energy per gram of the naturally occurring fuels.

Uncertainty of at least a factor of two to three is evident in estimates by different groups (International Atomic Energy Agency, World Energy Conference, US Energy Research and Development Administration) of the magnitude of US and world uranium resources available at costs of \$ 66/kg U_3O_8 or less. Uncertainties in the intermediate cost range of \$ 66/kg to perhaps \$ 250/kg are even larger. These uncertainties are significant in the context of estimating how long nations could rely on non-breeder reactors to supply a significant part of their energy use, but the uncertainties are *not* significant in the context of the energy potential of fission breeder reactors. This is so because: *(a)* the breeder's high energy extraction per gram of fuel stretches even the smallest estimated quantities of low-cost uranium out to 2000 TW·yr of electricity (1 TW = 10^{12} W = 1,000,000 MW); and *(b)* the insensitivity of electricity cost to fuel cost in the breeder means that high-cost, dilute uranium resources, which exist in quantities far larger than the low-cost resources, become economically acceptable as breeder fuel (the supply in the oceans alone represents a quantity on the order of 2,000,000 TW·yr of electricity). With fission breeders, therefore, nuclear fuel supply can be considered inexhaustible far beyond any time scale of conceivable planning interest.

The situation is similar for lithium, the limiting fuel resource for D-T fusion. There are uncertainties of a factor of three in US resources available at low cost ($ 60/kg of Li metal or less), and still larger uncertainties about world resources, but these uncertainties are essentially irrelevant to the outlook for D-T fusion. The *smallest* estimated global quantity of low-cost lithium represents at least 2500 TW•yr of electricity in D-T fusion reactors. Much more expensive lithium could be used economically, including presumably that in sea-water, which alone represents at least 100 million TW•yr of electricity. There is no reason to suppose then that lithium resources will limit D-T fusion on any interesting time scale. Deuterium in seawater (by weight) is 200 times more abundant than lithium, and is more economic to extract.

For the fission breeder, the high utilization of ores is reflected in a very low fraction that ore costs contribute to the total busbar cost. These fractions are a few tenths of a percent. This not only makes the low grade ores accessible, but it also provides for a different class of security for ore supply because such low grade ores can be found almost every-where, and storage of ores does not impose an unacceptable economical burden on the owner of a power plant. Therefore, ore supply embargoes are eliminated and, indeed, the operation of power plants is essentially decoupled from the traditional problems of fuel supply. For the fusion breeder, the situation is fundamentally the same as far as lithium in its function as a fuel is concerned. Indeed, the burn-up ore costs for both breeder types are on the order of 10^{-2} mill/kWh. But, for the case of the fusion breeder, lithium may have the additional function of serving as a coolant, which would raise the total inventory ore costs to a few tenths of a mill/kWh. In any event, for both breeder types, the ore costs are so low that the large existing uncertainties in ore prices that reflect geological and political conditions simply do not matter.

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APPENDIX TO CHAPTER II

II-A FACTORS EFFECTING THE ENERGY CONTENT OF NATURAL LITHIUM IN D-T FUSION REACTORS

(G.L. Kulcinski)

The energy content of a gram of natural lithium used in a D-T fusion reactor is not a simple quantity to arrive at; it depends on the form of lithium, the amount and type of structural material, and the strategy that is used for the management of this resource. We can make an attempt to estimate what the energy content might be by examining the various reactor designs that have been published in the literature. However, one must always keep in mind that, generally, these designs have *not* been optimized to get the greatest energy release per gram of Li.

For the purposes of discussion here, we will assume that tritium has an infinite half-life (this is the same as saying that tritium is rapidly transferred between plants so as to make full use of its energy content), and that no tritium atoms escape the plant. These assumptions will tend to make the energy content values higher than might be realized in practice. We will also assume that we have a steady-state fusion economy in operation, that is, no further expansion of the electrical generating capacity is occurring. This means that every tritium atom procuced can be burned and does not have to be used for plant inventory in new facilities. This again will tend to make the energy content values higher than might be realized in a fusion economy which is expanding at a finite rate.

We will consider the general case of the primary breeder reactor, containing both Li6 and Li7 (where the BR > 1), feeding one of two secondary types of reactors:

- near-breeders (utilizing primarily Li7 as a breeder, where the breeding ratio is less than 1 but more than 0.5);
- burner-reactors (containing no Li, and where no further breeding of tritium is performed).

In reality, one might have a mixture of both types of secondary reactors, because of environmental concerns over large tritium inventories in the center of cities and the desire to extend the fuel resources.

Examination of this general problem reveals that the energy content of a gram of natural Li is a function of four parameters:

(1) the total energy released (including blanket multiplication and breeding) per D-T plasma reaction in the ith type of reactor (i.e. primary or secondary) (E_R^i);

(2) the breeding ratio (BR);

(3) the ratio of tritium produced from Li6 to the total tritium produced (R);

(4) the number of tritium atoms that can be produced per D-T neutron by Li7 in a blanket concept (P).

The general relationship between the energy content of Li and the reactor parameters is shown in Figure II-A-1.

The general equation is then:

$$\frac{\text{kWh(th)}}{\text{g nat.Li}} = \underbrace{\frac{E_R^{Breed} \cdot A}{BR \cdot R}}_{\substack{\text{Breeder} \\ \text{Reactor}}} + \frac{(BR-1)\cdot A}{BR \cdot R} \cdot E_R^{Sec} \cdot \left[1 + \sum_{n=1}^{\infty} P^n\right] \qquad (1)$$

From Burning of Excess T from Breeder in Secondary Reactor

From Burning T Atoms in Secondary Reactors which were Bred in Secondary Reactor

Since P < 1 in all cases, we can rewrite:

$$\sum_{n=1}^{\infty} P^n = \frac{P}{1-P} \quad .$$

Then equation (1) becomes:

$$\frac{kWh(th)}{g\ nat.Li} = \frac{A}{BR \cdot R}\left[E_R^{Breed} + (BR-1)\cdot E_R^{Sec}\cdot\left(\frac{1}{1-P}\right)\right] , \qquad (2)$$

where:

$$A = \left(\frac{6\cdot 10^{23}\ Atoms\ Li6/mole}{6\ g\ Li6/mole}\right) \times \left(\frac{0.0742\cdot 6\ g\ Li6}{0.0742\cdot 6\ g\ Li6 + 0.9258\cdot 7\ g\ Li7}\right)$$

$$\times \left(\frac{1.6\cdot 10^{-13}\ Wsec}{MeV} \cdot \frac{1\ kWh}{3.6\cdot 10^{6}\ Wsec}\right) = 286\ \frac{Li6\ Atoms}{g\ nat.Li} \cdot \frac{kWh}{MeV} \quad .$$

Typical values for tritium breeding in various systems are given in Table II-A-I.

From Table II-A-1 we can see what range of values one might use for E_R, BR, R, and P. For four different liquid-Li-cooled reactors reported in the literature which utilized steel, Nb, and Mo alloys as structures, we find that the total energy released per D-T reaction is an average of approximately 20 MeV. This increases to approximately 21 MeV when Be is used with liquid lithium, but some of that energy increase is due to the Be. Finally, the use of solid-lithium breeding compounds and Be can raise the value of E_R to approximately 22 MeV (again approximately ten per cent of that increase is due to the Be). Therefore, it appears that a reasonable value for most fusion reactors is 20 MeV per D-T reaction. This value is not expected to change much if the Li6 is replaced with an absorber like

$$\frac{\text{kWh(th)}}{\text{g nat.Li}} = \left(\frac{\text{Energy Released in Breeder Blanket}}{\text{D-T Reaction}} \times \frac{\text{D-T Reactions}}{\text{T Atom Produced in Breeder Blanket}} \times \frac{\text{T Atoms Produced in Breeder Blanket}}{\text{Li Atom Burned in Breeder Blanket}} \times \frac{\text{Li Atoms Burned in Breeder Blanket}}{\text{Li6 Atom Burned in Breeder Blanket}} \times \left(\frac{\text{Li6 Atoms}}{\text{g Li6}} \times \frac{\text{g Li6}}{\text{g nat.Li}} \times \frac{\text{kWh(th)}}{\text{(MeV)}} \right) \right) +$$

$$\left(E_R^{Breed} \times \frac{1}{BR} \times 1 \times \frac{1}{R} \times A \right) +$$

$$+ \left(\frac{\text{Excess T Atoms Produced in Primary Reactor to be Burned in Secondary Reactor}}{\text{g of nat.Li Used in Breeder (Primary) Reactor}} \times \frac{\text{Total Energy Released in Secondary Reactor}}{\text{T Atom Burned in Secondary Reactor}} \right) + \left(\frac{\text{Tritium Atoms Produced in Secondary Reactor}}{\text{Excess Tritium Atom from Primary Reactor Burned in Secondary Reactor}} \times \frac{\text{Energy Released in Secondary Reactor}}{\text{T Atom Burned in Secondary Reactor}} \times \frac{\text{Excess T Atoms from Primary Reactor Which are Burned in Secondary Reactor}}{\text{g nat.Li}} \right) +$$

$$+ \left((BR-1) \cdot \frac{A}{BR \cdot R} \times E_R^{Sec} \cdot \frac{A}{A'} \right) + \left(P \times E_R^{Sec} \cdot \frac{A}{A'} \times \frac{(BR-1)}{BR} \cdot \frac{A'}{R} \right) +$$

$$+ \ldots + \left(\frac{\text{T Atoms Produced in Secondary Reactor from Burning Previous (n-1) T Atom Bred in a Secondary Reactor}}{\text{T Atom from Primary Reactor Burned in Secondary Reactor}} \times \frac{\text{Energy Released in Secondary Reactor}}{\text{T Atom Burned in Secondary Reactor}} \times \frac{\text{Excess T Atoms Produced in Primary Reactor to be Burned in Secondary Reactor}}{\text{g nat.Li in Primary Reactor}} \right).$$

$$+ \ldots + \left(P^n \times E_R^{Sec} \cdot \frac{A}{A'} \times \frac{(BR-1)}{BR} \cdot \frac{A'}{A} \right).$$

$$\text{Where } A' = \left(\frac{\text{Li6 Atom}}{\text{g Li6}} \times \frac{\text{g Li6}}{\text{g nat.Li}} \right).$$

Figure II-A-1: Relationship Between Energy Content of Lithium and Reactor Parameters

Table II-A-I: Summary of Selected Neutronic Characteristics of Various Fusion Reactor Designs

Design	Breeder	Fraction Li6	BR	$\frac{R}{T_6/T_{Total}}$	P[b]	Structure	E_R (MeV)
	Liquid Li (No Neutron Multiplier)						
UWMAK-I	Li	0.0742	1.49	0.59	0.60	SS	20.1
UWMAK-III	Li	0.0742	1.25[a]	0.58	0.52	TZM	21.1
ORNL	Li	0.0742	1.44	0.59	0.59	Nb	19
LLL	Li	0.04	1.36	0.63	0.50	SS	19.2
	Liquid Li + Be						
PPPL	Flibe	0.0742	1.07	0.87	0.13	PE16	21.4
LASL	Li + Be	0.0742	0.70	0.72		Nb	20.5
		0.90	0.34	1.0			
	Solid Li Compounds + Be						
UWMAK-II	$LiAlO_2$+Be	0.9	1.18	0.99	0.003	SS	21.6
BNL	LiAl+Be	0.9	1.48	1.0	0.0023	SAP	22.6
			average (nat.Li, no Be) = 19.9 MeV				
			average (liquid Li + Be) = 21.0 MeV				
			average $\left[\left(\begin{matrix}\text{solid enriched} \\ \text{Li6 compounds}\end{matrix}\right) + \text{Be}\right]$ = 21.1 MeV				

[a] Breeding on outside of torus only;

[b] T atoms produced per D-T neutron in Li7.

boron 10, because although the exothermic Li6 (n,T) reaction is no longer present, the exothermic B10 (n,α) and the (n,γ) reactions in the structure will compensate for this loss. A first approximation to E_R^N burner might also be approximately 20 MeV per D-T reaction. A similar argument might be made for a near-breeder reactor, where the endothermic Li7 (n,nT) reaction will be partially offset by increased exothermic reactions in the structure. Again for simplicity, $E_R^{Near\text{-}Breeder}$ = 20 MeV per D-T reaction is a reasonable approximation.

Next, one notes that, R, the ratio of tritium produced from Li6 to the total T production, is remarkably constant at 0.6 for the various liquid-lithium systems considered. This value jumps to 1 in the enriched system, where essentially all of the tritium comes from Li6.

The breeding ratio is normally quite high in liquid-lithium systems, ranging from 1.36 to 1.49 in early designs, which have been studied with uniform isotropic neutron sources completely enclosed by slab or cylindrical geometry. When toroidal neutronic effects are included along with the loss of neutrons in the divertor slots, beam tubes, and parts of the reactor, that are generally inaccessible, breeding ratios of 1.2 to 1.3 are more likely in liquid-lithium systems. It is expected that the proper treatment of neutron losses and peaked neutron fluxes typical of non-circular, non-uniform, and toroidal neutron sources will produce much lower breeding ratios than previously calculated in liquid-lithium systems.

The BR values for solid Li compound breeders are generally quite low (approximately one in most cases), and one is forced to use a neutron multiplier to get a breeding ratio greater than one. This may even be true with materials of high lithium density, such as Li_2O and Li_7Pb_2, when one includes the 10 to 20 per cent losses due to penetrations and inaccessible parts of the blanket as well as satisfying low neutron leakage to magnets. The effects of structural material are also particularly important with solid breeders, and they must be properly included in the calculation of overall breeding ratios. It is not satisfactory to simply calculate what BR one can get with complete enclosure of the neutron source and no significant parasitic capture in the structural material that is required to contain the high pressure gas usually used to cool solid-breeder systems.

The effect of adding non-breeding elements (oxygen, silicon, Pb) is to soften the spectrum such that Li7 reactions are reduced and the value of P is lowered, further reducing the BR.

Finally, the value of P is quite important in the overall fuel management strategy. If one were able to burn only the Li6 in Li then, as we will see later, the maximum energy content of one gram of natural lithium is approximately 6000 kWh (th)/g Li. Utilizing the Li7 reactions that would occur simultaneously in a liquid-Li system would raise this value to approximately 10,000 kWh(th)/g. However, this would leave roughly 0.9 gram of Li7 left over for every gram of natural Li originally in the reactor. One could still get some energy out of this excess Li7 if it was bombarded with D-T neutrons from the excess T produced in the fusion breeder reactor. In this case, one would want to keep the neutron spectrum as hard as possible in order to maximize the T_7 production. Calculations show that in liquid-Li7 systems, the number of T atoms produced per incident D-T neutron is approximately 0.50 to 0.6 (Table II-A-I). Obviously, one could do worse if the Li7 were in a solid compound with other non-fuel elements, or if a large fraction of neutrons escaped down penetrations in the blanket. It seems reasonable that when the toroidal neutronics and reasonable leakage is included, P is approximately 0.5 for liquid-Li7 systems (i.e., UWMAK-III in Table II-A-I). It is obvious that to use a solid breeder would lower P significantly and thus reduce the energy content of our lithium reserves.

Going back to equation (2) and assuming that as a first approximation,

$$E_R^{\text{Breed}} \simeq E_R^{\text{Secondary}} \simeq E_R^{\text{Primary}} ,$$

we find

$$\frac{\text{kWh(th)}}{\text{g nat.Li}} = \frac{AE_R}{BR \cdot R} \left[1 + (BR-1) \cdot \left(\frac{1}{1-P}\right)\right];$$

with our assumptions about

$$E_R \simeq 20 \text{ MeV/D-T Reaction} ,$$

and

$$R = 0.6 ,$$

we find

$$\frac{AE_R}{R} = 9533 \frac{\text{kWh}}{\text{g nat.Li}} .$$

If we allow the breeding ratio to be greater than one, then the energy content of the lithium in the primary breeder reactor is reduced, but this is actually more than compensated for if we burn the excess tritium in reactors containing the depleted Li from the breeder reactors. At a value of P = 0.5 we find that total energy content increases from 9533 kWh(th)/g nat. Li at BR = 1 to ≈ 12,000 kWh(th)/g nat. Li at an initial BR = 1.3 and 13,620 kWh(th)/g nat. Li at an initial BR = 1.75 (see Figure II-A-2). Since, as we previously said, breeding ratios of approximately 1.3 are probably the most likely maximum value for operating CTRs using liquid Li, a reasonable value of the energy content is approximately 9500 kWh(th)/g of natural Li if used in a breeder-burner combination, and it is approximately 12,000 kWh(th)/g of natural Li if used in a breeder/near-breeder combination. This latter scheme represents a 20 per

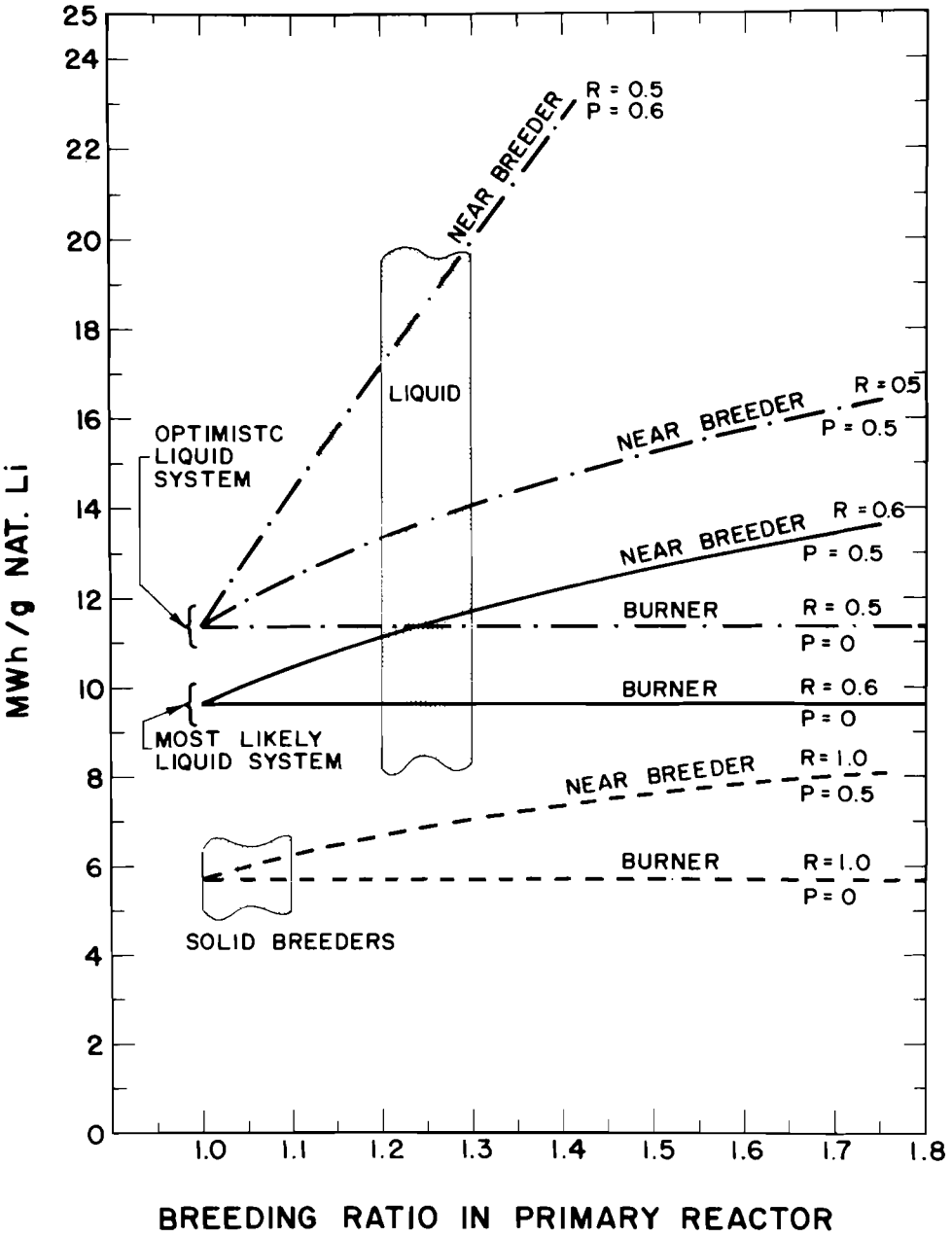

Figure II-A-II: Energy Content of Natural Lithium in Various Fuel Management Schemes For Fusion

cent increase in fuel reserves compared to the breeder/burner combination. Future generations will have to decide whether this expansion of fuel reserves is worthwhile versus having some reactors which have lower tritium inventories than the near-breeder reactors.

Other comments might be made about Figure II-A-2. For example, using lithium in solid-breeders (and discounting the energy contribution of Be which could be obtained in all configurations) we find that in the most probably BR range (1.0 to 1.1) the energy content of one gram of Li varies from 5700 to 6300 kWh(th)/g nat. Li if it is coupled with liquid-Li7-cooled near-breeders. However, it is not certain whether such a combination makes sense when the object of solid-breeders in the first place is to remove the difficulties associated with a liquid-metal system and to reduce the large tritium inventories associated with liquid-Li systems. If one used Li7 in solid form in the near-breeders very little benefit would be derived, and such a scheme hardly seems worth the effort.

Finally, if one could find a way of increasing the T_7 fraction and still maintaining high BR in breeder reactors, then the energy content could also be extended. This can only be done by reducing the energy degradation of the D-T neutrons by non-fuel atoms (i.e., structure, coolant, or alloying elements). It is unlikely that this number can be increased much above 0.7 in a realistic system, and it could easily drop as low as 0.4. We have included a curve for R = 0.5 and P = 0.6, which shows that at a BR of 1.3 the energy content could be improved approximately 60 per cent over the most likely case of R = 0.6 and P = 0.5.

In summary, if one has to specify the energy content of natural Li used in fusion reactors, it is most likely to be in the 12,000 kWh(th)/g range in a breeder/near-breeder scenario. If a breeder/burner scenario is used, then the energy content is approximately 9500 kWh(th)/g and in a solid-breeder/burner scenario a value of 6000 kWh(th)/g of natural lithium is most likely.

III. PRESENT STATUS OF FISSION AND FUSION REACTORS

G. Kessler, G.L. Kulcinski

1. FISSION REACTORS*

1.1 Introduction

The principle of breeding has been recognized from the very beginning of nuclear reactor development. Realizing that the η value--which characterizes the average number of neutrons produced by fission per absorbed neutron [(n,γ) + (n,f)]--is high for fast neutrons inducing the fission process, E. Fermi and W.H. Zinn began to design a fast breeder reactor as early as in 1944 [III-2]. This started the first round of fast breeder development, lasting from 1944 until roughly 1960, leading to reactors that are often referred to as fast breeders of the first generation. The US reactors EBR-I, EBR-II, and EFFBR, the British DFR, and the Russian BR-5 are the more prominent ones among them (see Table III-I) [III-3 to III-8].

Consistent with the general approach to reactor technology of these early years, the principal fuel was metal, more specifically U metal [III-9]; interconnected to that was the choice of Na as the principal coolant. The cores were small, the Na temperatures modest, and the breeding took place in the reflecting blanket and not so much in the core itself. With respect to long-term reactor strategies most attention was given to the doubling time [III-9, III-10]; the core inventory and fuel cycle costs were not so much in the focus of interest. All of these factors had certain consequences upon the way in which the problems were attacked.

Around 1960, with the background of a maturing thermal reactor technology, economic considerations developed which caused attention to shift to the nuclear fuel cycle as a whole [III-11]. In particular, it became clear that the burn-up of the fuel must be increased in order to achieve economic feasibility of fast breeder reactors. A light water reactor, for instance,

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* The following two subsections are a shortened and updated presentation of an article on Fast Breeder Reactors published in the *Annual Review of Nuclear Science* in 1970 [III-1].

Table III-I: First Generation Fast Breeder Reactors

		USA				USSR			UK
		CLEMENTINE	EBR-I	EBR-II	EFFBR	BR-1	BR-2	BR-5	DFR
Reactor Power									
Thermal	MW(th)	0.025	1.2	62.5	200	0	0.1	5	72
Electrical	MW(e)	0	0.2	20	66	0	0	0	15
Core									
Fuel		Pu metal	U metal	U metal	U metal	Pu metal	Pu metal	PuO_2	U metal
Core Volume	liter	2.5	6.0	65	420	1.7	1.7	17	120
Fuel-rating Average	MW(th)/kg fiss	0.0016	0.02	0.3	0.37	0	0.008	0.1	0.24
Power-density Average	MW(th)/liter	0.01	0.17	0.8	0.45	0	0.06	0.3	0.5
Linear Rod-power Maximum	W/cm	(av 50)	300	450	250	0	150	200	(av 320)
Neutron-flux Maximum	$n/cm^2 \cdot sec$	(av $5 \cdot 10^{12}$)	$1.1 \cdot 10^{14}$	$3.7 \cdot 10^{15}$	$4.7 \cdot 10^{15}$	$5 \cdot 10^{10}$	$1 \cdot 10^{14}$	$1 \cdot 10^{15}$	$2.5 \cdot 10^{15}$
Primary Heat-transfer System									
Coolant		Hg	NaK	Na	Na	-	Hg	Na	NaK
Coolant Temperature at									
Core inlet	°C	40	230	370	290	-	30	375	200
Core outlet	°C	120	320	470	430	-	60	450(500)	350
Coolant Mass Flow	m^3/h	0.6	80	2200	5500	-	6	240	1800
Number of Coolant Loops		1	1	2	3	-	1	2	24
Time Schedule									
Design		1945	1945					1956	
Construction		9/1946	1949	1957	8/1956			1957	3/1955
First Criticality		11/1946	8/1951	10/1961	8/1963			6/1958	11/1959
Full Operation		3/1949	12/1951	4/1965	8/1966	1955	1956	7/1959	7/1963
Shutdown		6/1953	1963			1956	1957		
Remarks		First fast reactor First Pu-fueled reactor	First nuclear electricity generation, Pu-core since 1962	Reactor plant with integral fuel processing facility	Since October 1966 out of operation			UC-core since 1965	

requires a burn-up of only 25,000 to 35,000 MW(th)·day/t in order to burn effectively all original fissionable atoms. Fast reactors inherently require high enrichment, and burn-ups of ≈100,000 MW(th)·day/t are required in order to keep the number of passes of an individual fissile atom through the full fuel cycle to a tolerable level. To achieve these required high burn-ups, it was mainly UO_2/PuO_2 which offered the best chance (see Chapter IX).

This led to the so-called second generation of fast breeders. The paper by J.B. Sampson and E.A. Luebke [III-12] was the first step. After the IAEA seminar on the Physics of Fast and Intermediate Reactors in Vienna in 1961 [III-13], this ceramic fast-reactor scheme received worldwide attention together with the shift of emphasis from breeding to economy [III-14 to III-16].

It became apparent that the reactor design would differ somewhat from that of the first generation, because of neutron moderation by the oxygen atoms in the UO_2/PuO_2 fuel. At first, then, most effort concentrated on calculating the Doppler coefficient [III-17, III-18]. Major undertakings such as the SEFOR reactor resulted [III-19, III-20]: it was specifically designed to measure and demonstrate the Doppler coefficient for a variety of fast-reactor transients. The possibilities of measuring the Doppler coefficient in a critical zero-power facility were also explored, and finally understood in the early sixties.

After it had been realized that the Doppler coefficient would be of sufficient negative magnitude, interest turned to the sodium-void effect [III-21]--the reactivity change that occurs after the core has been fully or partly voided of sodium. It was found to be positive for sufficiently large fast cores [III-18], which caused major concern in the fast-reactor community as of 1964. The four large 1000 MW(e) design studies of General Electric, Westinghouse, Combustion Engineering, and Babcock and Wilcox greatly concentrated on this problem [III-22]. Among other things, these studies revealed that it was too restrictive to demand that all power coefficients be negative.

Related to these problems is the question of the target size of a reference fast reactor. In the first fast-reactor generation the considered sizes were very small, up to 66 MW(e) (EFFBR), but it became clear in the early sixties that much larger reactor stations had to be envisaged. Between 1959 and 1963, 500 MW(e) was often considered a good target size [III-23], but after the above-mentioned four design studies in the US and other studies [III-24], 1000 MW(e) was generally accepted as a target value for realistic fast reactor designs.

With the power coefficients these questions are interrelated to inherent fast-reactor safety. At the Argonne Conference of 1965, the various then existing fast-reactor reference designs [III-25] were analyzed with respect to possible chains of events that could lead to a major accident. It became apparent that the Na-void effect becomes important only if a very unlikely

type of major accident takes place, for instance the malfunction of the shut-off systems followed by Na-boiling effects. Along the same lines it was recognized that a series of other phenomena require the same attention as the Na-void effect, for instance Na-boiling and superheating, as this may be responsible for the time scale of Na-voiding [III-26 to III-28]. These considerations were the starting point of out-of-pile and in-pile safety research programs carried out between 1968 and 1976, which laid the basis for licensing prototype power reactors of the 300 MW(e) class.

In 1966/1967 the first results of high burn-up pin irradiations of ≈70,000 to 80,000 MW(th)·day/t were obtained in thermal reactors such as GETR at Vallecitos and others, and pins followed which had been irradiated under fast-reactor conditions, particularly at DFR and EBR-II. They all indicated good results. From that it was generally concluded that 50,000 MW(th)·day/t would be a good starting value for the performance of a first fast core of a prototype. So most fast-reactor groups of the world decided as a first step to build 300 MW(e) Na-cooled prototypes. However, long-range strategic considerations had revealed the desirability of having 1000 MW(e) fast breeder power stations by 1980 to 1985 [III-29, III-30].

Another fairly strong argument was forwarded by K.P. Cohen [III-31] and others: The unexpected large-scale installation of light-water reactors (LWR) leads to a large production of Pu. If Pu is being recycled into these LWRs, its value as compared to that of U235 is only something like 80%, while if introduced into a fast reactor the Pu value is of roughly 140% [III-32]. In other words, fast reactors can stand a high Pu price and, therefore, there is a natural partnership between LWR and fast breeder.

Along these two lines of argument, the UK was the first in the West to go ahead with the 250 MW(e) plutonium-fueled fast reactor PFR at Dounreay [III-33], France pushed the design and construction of the 250 MW(e) PHENIX prototype [III-34], and the FRG together with Belgium and the Netherlands started work for their 300 MW(e) SNR reactor [III-35]. In the USSR, the design and construction of the BN 350 [III-36] was going on and, at least timewise, this Russian group was in the lead for this class of prototype reactors. The US and Japan will follow with the construction of CRBR [III-37] and MONJU [III-38] around 1977 and 1978, respectively. In Table III-II more details are given for these prototype reactors, or, as they are called in the US, demonstration reactors.

Sodium technology must be duly mastered and cheap enough if the utilities all over the world are to rely on it. The question of reliability, availability, and capital cost of these Na components is, therefore, under constant discussion and investigation. Problems of irradiation damages under high neutron fluences ($>10^{23}$ n/cm^2) led to the US decision to press more for the large-scale fast-neutron test-reactor FFTF in order to have an orderly procedure in the design and development of large fast breeders, even if this leads to some time delay [III-39, III-40].

Table III-II: Fast Breeder Prototype and Demonstration Reactors

	UK		FRANCE		FRG	USSR		USA	JAPAN
	PFR	CFR-1	PHENIX	SUPER-PHENIX	SNR 300	BN 350	BN 600	CRBR	MONJU
Reactor Power									
Thermal MW(th)	600	2900	563	2910	736	1000	1480	950	714
Electrical MW(e)	270(254)	1320(1250)	250	1200	312(282)	350[a]	600	360	300
Primary Circuit	Pool	Pool	Pool	Pool	Loop	Loop	Pool	Loop	Loop
Number of Loops					(3)	(6)[b]		(3 or 4)	(3)
Diameter of Reactor Vessel m	12.2	22.5	11.8	≈21	6.7	6.0	12.8	6.2	7.1
Coolant	Na	Na	Na	Na	Na	Na	Na	Na	Na
Coolant Temperature at									
Core inlet °C	400	400	400	395	377	300	380	387	390
Core outlet °C	562	562[c]	560	535	546	500	550	540	540
Core Dimensions									
Height cm	91	100	85	100	95	106	75	91	90
Diameter cm	147	≈290	139	≈366	178	158	205	108	178
Fuel	UO_2/PuO_2	UO_2/PuO_2	UO_2/PuO_2[d]	UO_2/PuO_2	UO_2/PuO_2	UO_2	UO_2/PuO_2[e]	UO_2/PuO_2	UO_2/PuO_2
Cladding	SS	SS	SS	SS	SS(1.4970)	SS	SS	SS 316 20% c.w.	SS
Pin Diameter mm	5.84	5.84	6.6	8.65	6.0	6.1	6.9	5.84	6.5
Number of Fuel Pins per Fuel Element	325	325	271	271	169	169	127	271	169
Maximum Rod Power W/cm	450	450	430	450	460	470	530	475	457
Maximum Clad Temperature (hot spot) °C	700	700[c]	700	700	670	660	700	640	700
Maximum Fluence at Clad n/cm²			$3 \cdot 10^{23}$	$4.6 \cdot 10^{23}$	2 to $3 \cdot 10^{23}$			$3 \cdot 10^{23}$	
Burn-up MW(th)·day/t	70,000	>70,000	50,000	70,000	55,000	50,000	90,000		80,000
Breeding Ratio	1.2	≈1.2	1.16	1.17	1.0-1.2	1.0-1.4	0.9-1.3	1.23	1.2
Status	In operation since 1975	Planned, start of construction 1978	In operation since 1973	Planned, start of construction 1976	Start of construction 1973 Start of operation 1982	In operation since 1973	Under construction, start of operation 1977	Planned, start of construction 1977	Planned, start of construction 1978
Site	Dounreay (Scotland)		Marcoule (France)	Creys-Malville	Kalkar (FRG)	Shevshenko (Caspian Sea)	Bjelojarsk (Ural)	John Sevier (Tenn.)	
Supplier	TNPG	TNPG[f]	CEA		SNR Consortium				PNC
Operator	UKAEA	CEGB	CEA/EdF	EdF/ENEL/SBK	SBK	State Committee for Atomic Energy	Ministry for Electric Energy	TVA Commonwealth Edison	

[a] or 150 MW(e) + 120,000 m³/day fresh water

[b] including one as reserve

[c] preliminary values, may be lower

[d] initial loading partly with UO_2

[e] at first UO_2

[f] with SNR Consortium

Large fast-breeder reactors of 1000 MW(e) using a ceramic fuel instead of a metallic fuel also allow for coolants other than Na. The large size of 1000 MW(e) brings the fraction of fissile atoms down to approximately 15%, as compared to 35% or more in the first generation of fast breeders; this is *one* factor of two. The density of UO_2/PuO_2 fast-reactor fuel is small as compared to the density of metallic fuel in the first generation; it gives *another* factor of two. Therefore, the power density in such a large fast ceramic reactor is lower by at least a factor of four as compared to the early concept of the first generation. This explains in principle the additional degree of freedom one obtains with respect to the coolant. A number of groups thus considered He as a fast-reactor coolant [III-41, III-42]. But helium-cooled fast-breeder reactors also require excessive fuel-pin testing in a proper environment and therefore a proper test bed. This leads to the problem of a test reactor with He as a coolant, and introduces time delays as compared with the Na [III-43]. But it seems to be clear that such a He-cooled fast reactor has a long-range potential if the HTGR or thorium-converter line reaches its commercial application [III-44, III-45].

The overall picture in most cases shows a line of development that leads to prototype reactors of the 300 MW(e) class, all scheduled around 1970 to 1978 for the start of construction, and around 1973 to 1984 for completion. The UK, France, the USSR and the FRG together with Belgium and the Netherlands are already preparing for the next step with fast breeders of 1200 to 1500 MW(e) [III-46 to III-49]. Complementary to that is the US line of development with the FFTF being another major milestone of fast breeder development. To complete the picture it should be mentioned that Italy has decided to build PEC, a fairly small but versatile test reactor that should serve as a starting point for a possible future evolution [III-50]; in India an experimental fast reactor is under construction. Japan, on the other hand, has decided to make the development of a Na-cooled fast-breeder reactor a major national project. Their development is a few years behind the European developments, but it is a very well coordinated and fully self-consistent effort [III-51, III-52].

The major data of the second generation experimental fast reactors are given in Table III-III. This table, as well as tables III-I and III-II, illustrates the above broad picture.

It should be mentioned here that an alternative approach to breeding in a thermal reactor is being pursued at Oak Ridge, where the molten salt breeder reactor concept is under development using the thorium-U233 fuel cycle. The successful operation of an experimental reactor, the MSRE (Molten Salt Reactor Experiment) [III-53], which first became critical in 1965, demonstrated that the concept of a fuel dissolved in molten fluoride salts is feasible. The fuel is circulated during operation, and reprocessing is performed in a small on-site plant. In October 1969 the MSRE became the world's first reactor to operate on U233. This concept is still in an early stage of development; it is judged that it

Table III-III: Second-Generation Experimental Fast Reactors

		USA		USSR	France	Germany	Italy	Japan
		SEFOR	FFTF	BOR 60	RAPSODIE*	KNK-II	PEC	JOYO
Reactor Power								
Thermal	MW(th)	20	400	60	40	58	130	100
Electrical	MW(e)	0	0	12	0	20	0	0
Core								
Fuel		PuO_2/UO_2	PuO_2/UO_2	PuO_2/UO_2 or UO_2	PuO_2/UO_2	PuO_2/UO_2 + UO_2	UO_2	PuO_2/UO_2
Core Volume	liter	500	1030	53	45	320	420	280
Linear Rod-Power Maximum	W/cm	650	500	590	400	430	400	430
Neutron-Flux Maximum	$n/cm^2 \cdot sec$	$6 \cdot 10^{14}$	$7.2 \cdot 10^{15}$		$3 \cdot 10^{15}$	$2.3 \cdot 10^{15}$	$2.8 \cdot 10^{15}$	$4 \cdot 10^{15}$
Primary Heat-Transfer System								
Type		Loop	Loop		Loop	Loop	Loop	Loop
Coolant		Na	Na	Na	Na	Na	Na	Na
Number of Coolant Loops		1		2	2	2	2	2
Coolant Temperature at								
Core inlet	°C	370	320	360to450	410	360	375	370
Core outlet	°C	430	480	600	530	550	525	500
Date of Operation		1969	1978	1969	1970	1977		1977

* modified version

might become economically attractive by the end of the century.

Returning to the main line of fast breeders, we now will review the physics area of fast breeder development in more detail.

1.2 Fast Reactor Physics--Scientific Feasibility

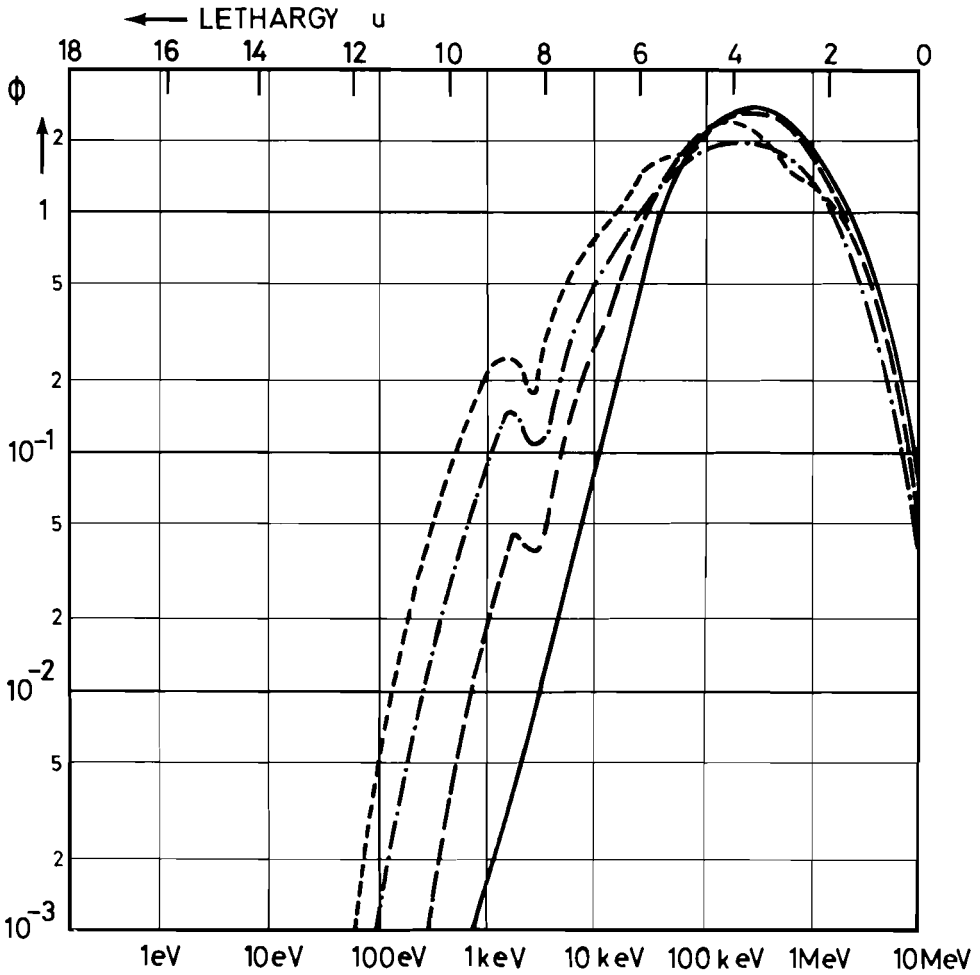
Among the main reasons for distinguishing between first- and second-generation fast breeders are their neutron spectra, and Doppler coefficients, and Na-void coefficients.

As mentioned in 1.1, the cores of the reactors of the first generation were small and fueled with metal, and their spectra were accordingly hard. The cores of the second generation are large and diluted and fueled with mixed oxides; thus their spectra are comparatively soft. This affects the flux intensity in the resonance region of the cross sections involved, and the design of the second generation of fast breeders must account for these resonance phenomena in detail. In Figure III-1 four spectra are plotted to exemplify this behavior: the effective fission cross section σ_f of Pu239 indicates the softness of the various spectra. In addition, the percentages of fission are compared in Table III-IV.

Table III-IV: Comparison of the Relative Fractions of Fission for Fast Breeder Reactors

Neutron Energy	EFFBR (First Generation Fast Breeder Reactor)	Reference Fast Breeder Reactor of the Second Generation
Below 9.1 keV	0.5%	9.6%
Below 40.7 keV	6.3%	25.2%
Below 67.0 keV	11.8%	35.5%

As these ceramic-fueled breeder reactors cannot rely on thermal fuel expansion as their main inherent stability feature, the Doppler coefficient must provide this stability. The fissionable isotopes give a positive contribution with temperature-resonance broadening, whereas the fertile isotopes have a negative effect, so that one must make sure of the sign and size of the Doppler coefficient of a given core composition. G. Goertzel [III-54] was the first to calculate the Doppler coefficient, but he concentrated on the 100 keV region and neglected the



		Spectral averaged σ_f of Pu239 (barn)
———	EFFBR	1.61
— —	1000 MW(e) Breeder Metal Fuel	1.61
—·—·—	1000 MW(e) Breeder Carbide Fuel	1.71
- - - -	1000 MW(e) Breeder Oxide Fuel	1.78

Figure III-1: Neutron-Flux Spectra of Fast Reactors

region of stronger and more isolated resonances. Later, R.B. Nicholson [III-55] calculated this coefficient by a group-wise procedure covering the entire energy spectrum. Then the groups of Argonne, Karlsruhe, General Electric [III-56 to III-58], and others extended and refined the calculations. Especially the treatment of overlap between resonances received much attention. Now the development of the theory is practically complete; survey papers on Doppler coefficient calculations were published by L.W. Nordheim [III-59] and by R.B. Nicholson and E.A. Fischer [III-60]. Besides the standard method of calculating effective cross sections, purely numerical methods were developed, which require high-speed digital computers.

For large ceramic reactors the Doppler coefficient is approximately proportional to 1/T. Typical for the so-called Doppler constant $T \cdot dk/dT$ (T is the temperature in ^0K, k the criticality factor) for typical power reactor designs are:

300 MW(e) Na-cooled prototype reactor: -0.0055 [III-61]

1250 MW(e) Na-cooled reactor: -0.008 [III-62].

The first measurements of the Doppler coefficient were carried out by W.Y. Kato and D.K. Butler [III-63] in a facility simulating the hard spectrum of EBR-I using cyclic heating of the samples. In the meantime quite a number of experiments have been performed with gradually increasing accuracy [III-64 to III-67]. For that it was necessary to relate the results of the measurements to reactor theory. This was first done by F. Storrer [III-68] and later by E.A. Fischer [III-69]. For U238 in Na-cooled assemblies, the analysis of an experiment in ZPR-VI,5 with ENDF/B data gave calculated values that are about 25% too low in magnitude [III-66]. Similar results were obtained in a recent analysis of the ZEBRA Doppler-loop measurements [III-70].

A completely different approach for measuring the Doppler coefficient is to use power excursions of a properly designed fast test reactor. The GODIVA assembly of Los Alamos [III-71] and its experiments for determining the metal fuel expansion coefficient influenced the conception of this approach. The groups of Karlsruhe and General Electric independently developed this plan and later combined their resources to build up the SEFOR project [III-20]. SEFOR was a reactor with a 500 liter sodium-cooled core, 20% enriched UO_2/PuO_2 rods with a neutron spectrum and fuel temperatures of future large LMFBRs [III-72]. Control was effected by a movable Ni reflector in order to have a clean core geometry. A fast-reactor excursion device allowed for the rapid expulsion of a central absorber, and thereby the introduction of reactivity ramp rates of up to 200 $/sec (1 $ is the reactivity where the reactor becomes prompt critical) with various reactivity values [III-73, III-74].

The SEFOR experimental program began in April 1969, and was successfully completed in January 1972. The program consisted of a series of static, oscillation, and transient tests (both

subprompt and super-prompt critical) [III-75 to III-78]. The central objective of the experimental program was to verify that the LMFBR has a prompt negative Doppler coefficient of sufficient magnitude to make it an extremely stable and inherently safe reactor system, and to obtain an accurate measurement of the coefficient. The super-prompt critical transient tests, in particular, involving reactivity insertions of up to 1.3 $ in 0.1 sec, provided a most convincing demonstration of the effectiveness of the Doppler coefficient in limiting the energy release in a fast reactor power excursion (see Figure III-2).

In the super-prompt critical transients, minimum periods of about 2 msec and peak powers of 10,000 MW(th) were achieved. As a result of known uncertainties in the basic reactor measurements (power and reactivity) and the good agreement between all tests, the experimental uncertainty on the measured $T \cdot dk/dT$ was estimated to be ± 12% [III-79]. Measured and extrapolated values and uncertainties of Doppler power coefficients C_D and Doppler energy coefficients γ_D are given in Table III-V.

Table III-V: Measured and Estimated Uncertainties for Fast-Reactor-Core Doppler Coefficients [III-79]

	Measured in SEFOR	Extrapolated for 1000 MW(e) LMFBR
Doppler Temperature Coefficient $T \cdot \frac{dk}{dT}$	± 14%	± 12%
Doppler Power Coefficient C_D	± 8%	± 15%
Doppler Energy Coefficient γ_D	± 10%	± 14%

Excursion tests to measure the Doppler coefficient were also pursued on VIPER in the UK [III-80, III-81].

The next big challenge to fast-reactor physics after the Doppler coefficient was the understanding and the calculation of the Na-void coefficient. Both the theoretical and the experimental treatment turned out to be difficult, as the effect is governed by differences of major effects. Removal of Na from a fast-reactor core affects the reactivity in three ways:

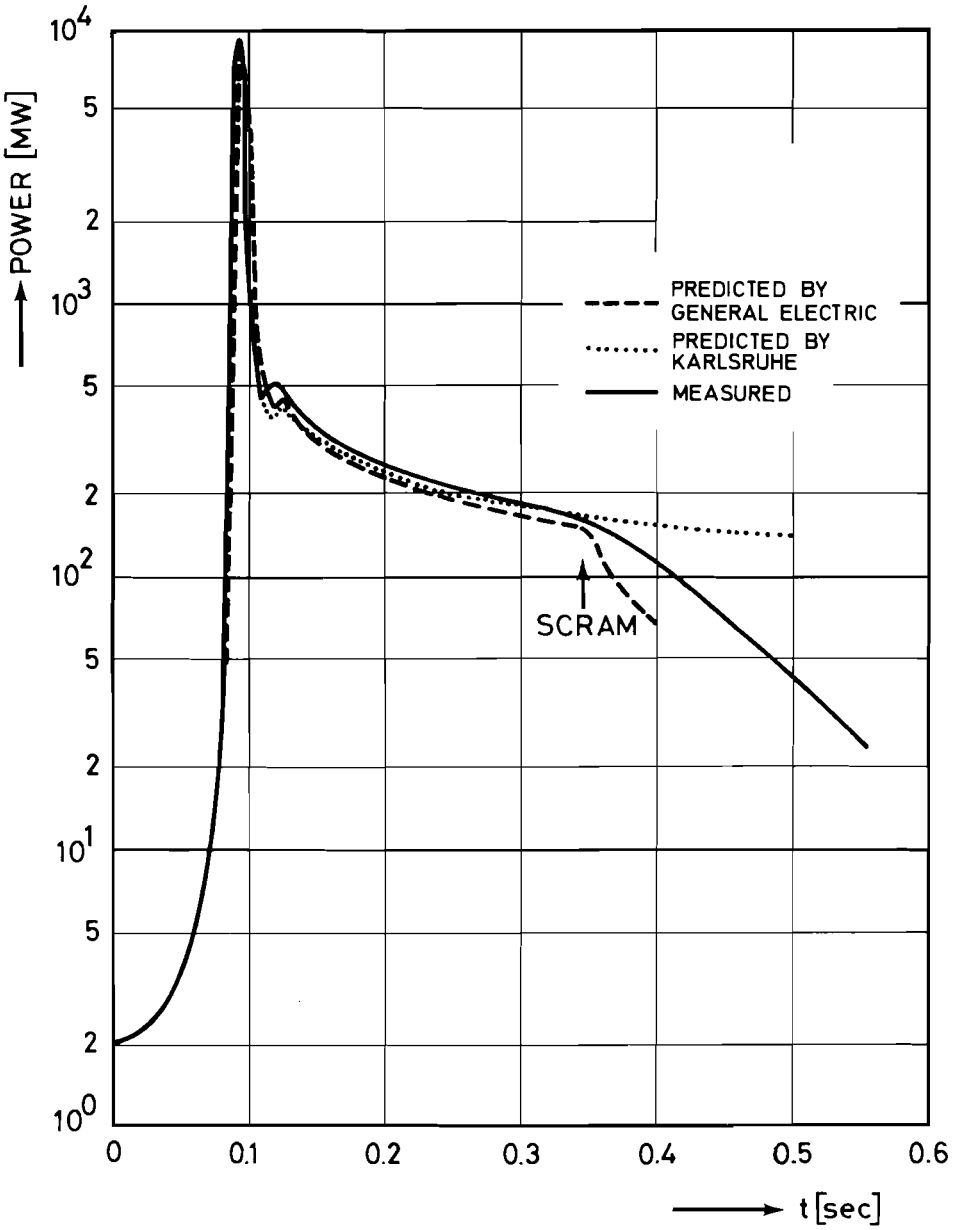


Figure III-2: SEFOR Core I Super-prompt Transient

(1) no more neutrons captured in Na (small positive effect);

(2) hardening of the spectrum; because of threshold fission in U238 and because η increases with energy, this is a positive effect in low-leakage reactors;

(3) increase of leakage (negative effect, dominates near core boundaries).

The Na-void coefficient is negative in breeder reactors of the first generation. It was first pointed out by J.B. Nims and P.F. Zweifel [III-21] in 1959 that the void effect might be positive in large ceramic reactors. The Na-void coefficient, as well as the multiplication factor k, is mostly calculated in diffusion theory by the multigroup method, where the range of energies of the neutrons present in the reactor is divided into a suitable number of intervals, or energy groups.

The principal tool of investigating the Na-void effect experimentally is the fast critical facility. A survey of existing facilities is given in Table III-VI. Measurements have been made at ZPR-III, ZPR-VI, ZEBRA, SNEAK, and elsewhere.

As already mentioned, the more complex physics of second-generation fast breeders requires a much improved calculational technique. Since the spectrum of the first-generation breeders was hard enough not to cover the resonance area, it was adequate simply to superpose the cross-section contributions of the various isotopes present in the core. Therefore, it was possible to establish universal sets of group constants. The mode of averaging these universal group constants within one energy group was fairly unsophisticated and simple [III-82]. But with improved calculational techniques and the investigation of the Doppler coefficient it became clear that the energy self-shielding must be taken into account. This implied that it was no longer possible to establish universal sets of group constants. It was necessary to have programs to prepare group constants for each particular reactor composition in question as an input for an extended multigroup calculation, e.g. the MC^2 program of Argonne [III-83], the GALAXY program of UKAEA [III-84], or the MIGROS program of Karlsruhe [III-85]. Now, very often a zero-dimensional or one-dimensional many-group calculation is run first (e.g. 60 groups or even more) and the resulting energy spectrum is used to prepare for each individual core or blanket composition a condensed individual set of (coarse) group constants which are finally used in a multi-dimensional calculation [III-86 to III-90].

The extremely refined calculational technique is, of course, very sensitive to the microscopic data input, while a tremendous input of microscopic cross-section data is needed to run the above-described sophisticated calculations. In order to get these microscopic input data, the European American Nuclear Data Committee (EANDC) compiled the various requests [III-91] and,

Table III-VI: Fast Critical Assemblies

	Location	Year of First Criticality	Short Description	Fissile Material	Typical Core Size (liters) (for average reflector thickness)
ZPR-III	Argonne, Id., USA	1955	Horizontal, split-table machine	U235, Pu239 (600 kg)	600
FCEL	Atomics International, Calif., USA	1960	Horizontal, split-table; thermal driver	U235, 25 kg of U233 were used in some assemblies	100 (test zone)
VERA	Aldermaston, UK	1961	Vertical, split-table	U235, Pu239 (40 kg)	400
BFS	Obninsk, USSR	1961	Vertical, fixed	U235	1800
ZEBRA	Winfrith, UK	1962	Vertical, fixed	U235, Pu239 (400 kg)	3000
ZPR-VI	Argonne, Ill., USA	1963	Horizontal, split-table	U235	3000
ZPR-IX	Argonne, Ill., USA	1964	Horizontal, split-table	U235	3000
FRO	Studsvik, Sweden	1964	Vertical, split-table	U235	65
MASURCA	Cadarache, France	1966	Vertical, fixed	U235, Pu239 (200 kg)	3000
SNEAK	Karlsruhe, FRG	1966	Vertical, fixed	U235, Pu239 (200 kg)	3000
FCA	Tokai-Mura, Japan	1967	Horizontal, split-table	U235, Pu239 planned	3000
ZPPR	Argonne, Id., USA	1968	Horizontal, split-table	U235 about 3000 kg of Pu239 planned	3000
STEK	Petten, Netherlands	1969	Vertical, fixed, thermal driver	U235	250

for example, the Cross Section Evaluation Working Group at Brookhaven has compiled an Evaluated Nuclear Data File for Reactor Applications (ENDF/B) [III-92, III-93]. The International Atomic Energy Agency has been making evaluations and coordinating on a worldwide basis for some time [III-94].

More general evaluations have been made to assess the influence of uncertainties in microscopic data on reactor variables. To be mentioned here is the paper of P. Greebler [III-95].

This consideration can perhaps be brought to an end best by stating that the prediction of k_{eff} is within 3% to 4% (critical mass: 15% to 20%), if a particular case is calculated and the group-constant set is chosen without care among the sets in use. If one instead chooses or prepares the group-constant set with care, but without special experimental assistance, the prediction of k_{eff} is within 1.5% to 2% (critical mass: 8% to 10%). If in addition to the latter case the results of a similar critical experiment are available, it should be possible to predict k_{eff} within 0.5% to 1% (critical mass: <5%) [III-96, III-97].

1.3 State of Development of Fast Reactors--Engineering Feasibility

Fast reactor cores can be cooled by liquid metals (Na, NaK or Hg) or high pressure gases (He, steam, etc.) [III-3, III-98]. In all cases the neutron spectrum remains relatively hard, which is important for breeding. Sodium as a coolant is clearly dominating the fast-reactor development scene. Between 1960 and 1970, H_2O-steam cooled and D_2O-steam cooled fast reactor concepts were studied in the USA and the FRG [III-99, III-100]. These design concepts, however, were abandoned for technical reasons around 1970.

Helium-cooled fast reactor design concepts and construction plans are still being followed as the alternative coolant concept to liquid-metal-cooled fast-breeder reactors [III-45]. In the following chapter we first deal with the technical aspects of sodium as a coolant.

1.3.1 Technical Aspects of Sodium-Cooled Fast Breeder Reactors

The LMFBR design concept is mainly determined by the choice of sodium as a coolant. Sodium

- has a high melting point (100°C) and a high boiling point (900-1000°C);
- has high specific heat and very good thermal conductivity;
- is opaque and becomes radioactive under neutron irradiation within the core;
- reacts chemically with water and air.

The high melting point of sodium requires preheating systems of pipes and components of the cooling circuits before operational conditions are reached at start-up. The high boiling point allows for high coolant-temperature conditions under very low system pressures (6 to 10 atm). This results in high thermal efficiencies for the overall LMFBR plant in the range of 40%. The relatively high specific heat permits moderate coolant velocities of 2 to 5 m/sec within the fuel elements and low pumping power, whereas the good thermal conductivity together with other thermal properties lead to extremely good natural convection conditions in the core and coolant system during shut-down and emergency core-cooling conditions. However, these excellent thermal properties also give rise to special design and operational consequences. Thermo-shock problems within the reactor vessel at pipe nozzles and valves must be avoided during short-term power reductions or reactor scram conditions.

Although sodium is not very corrosive against stainless steel, its impurities, mainly oxygen and carbon, must be held at very low contents by special cold traps in the bypass of the coolant circuits (5 to 10 ppm for O_2, <50 ppm for carbon) [III-101]. At high impurity contents, radioactive corrosion products dissolve from the surface of the fuel cladding. They are then transported to low-temperature parts of the primary coolant circuit (heat exchangers). This must be avoided because of maintenance and repair difficulties which could arise after several years of plant operation.

The opacity of sodium affects the design of the refueling systems and requires ultrasonic devices for the supervision of refueling and repair processes.

Major design consequences arise from the potential to chemical reactions of sodium with water and air. This property together with the fact that sodium becomes radioactive under neutron irradiation in the core leads to a plant design with

- a primary coolant circuit which contains the radioactive sodium heated up in the core;
- a secondary coolant circuit coupled with the primary one by intermediate heat exchangers;
- a tertiary water circuit producing steam for electricity generation by means of the turbine-generator system.

Within the primary coolant circuit radioactive sodium is protected against air by steel barriers and argon- or nitrogen-filled cells. Radioactive sodium of the primary coolant circuit is separated from the non-radioactive sodium of the secondary one by steel tubes of the intermediate heat exchangers.

So far two principal design concepts have been used for LMFBRs, the pool and the loop concepts (Figure III-3). With the pool concept the entire primary circuit components with core,

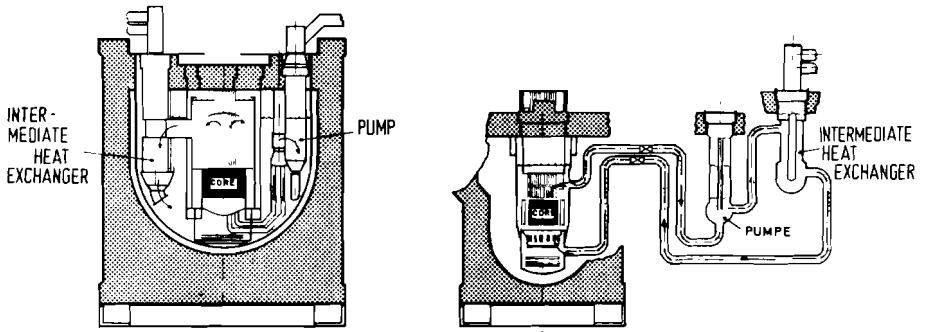


Figure III-3: Alternative LMFBR Design Concepts--Pool and Loop

primary pumps and intermediate heat exchangers are built into the sodium filled pool tank. This concept was used for PFR, CFR-1, PHENIX, SUPERPHENIX and BN 600 (see Table III-II). Figure III-4 shows the pool reactor system of the French 250 MW(e) reactor PHENIX. With the loop concept, in contrast, only the reactor core is built into the reactor vessel, and the primary sodium is pumped to the intermediate heat exchangers via a piping system. This design concept was chosen for BN 350, SNR 300, CRBR and MONJU (see Table III-II). Figure III-5 shows the loop reactor system of the US demonstration plant CRBR. Both design concepts have a number of advantages and disadvantages which roughly balance each other [III-102, III-103]. Only future experience with operation and licensing of large LMFBRs can perhaps show the more favorable design concept.

The reactor core is arranged on a grid plate that supports core fuel and radial blanket elements. The radial blanket elements are surrounded by reflector and shielding elements which serve to protect the wall of the reactor vessel against high neutron irradiation.

The reactor core (see Figure III-6) contains a number of hexagonal fuel and a few absorber elements, both arranged in two radial core zones of different fuel enrichment. Different radial core enrichment is usually applied in order to flatten the radial profile. Hexagonal fuel elements contain between 169 and 271 fuel pins, with core fuel (PuO_2/UO_2) in the middle and fertile fuel (U238) at both ends (see Figure III-7).

The fission gases produced within the fuel during irradiation are collected in a fission gas plenum at the end of each pin. The fission gas plenum can be arranged below the lower axial blanket zone or above the upper blanket zone. The absorber elements contain a similar arrangement of B_4C (boron-carbide) rods.

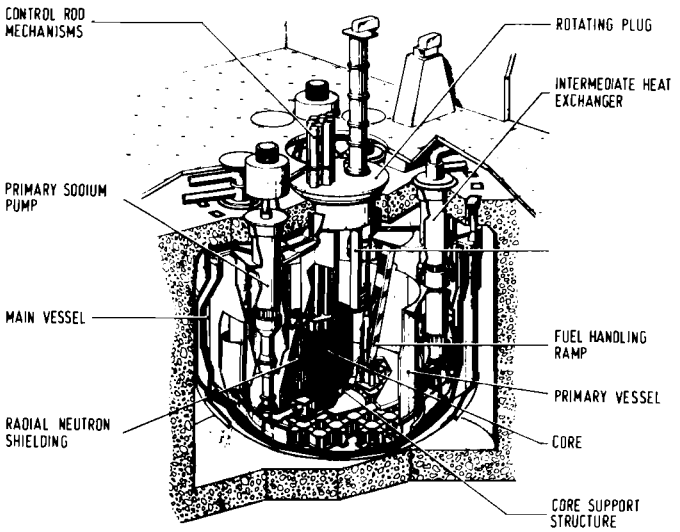


Figure III-4: French Fast Breeder Prototype Reactor PHENIX

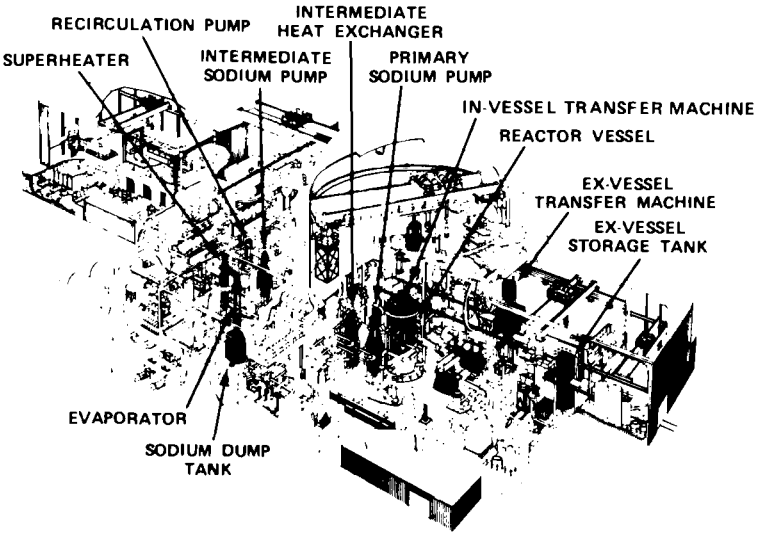


Figure III-5: US Fast Breeder Demonstration Plant CRBR

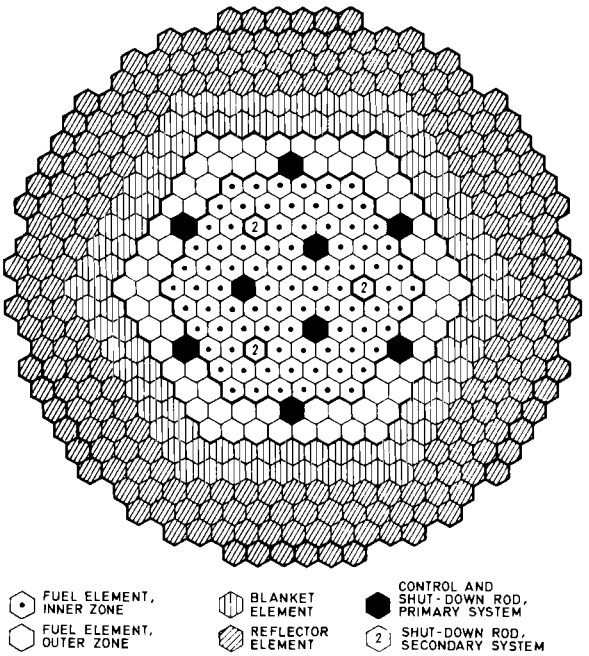


Figure III-6: Cross Section of SNR 300 Reactor Core

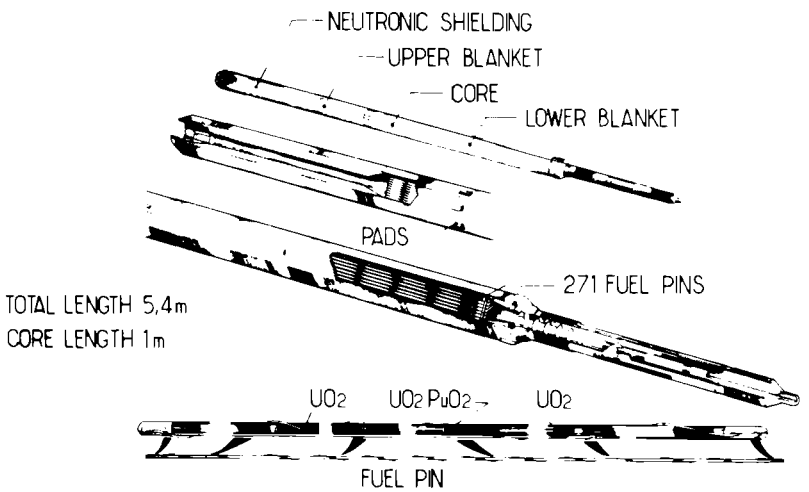


Figure III-7: Fast Breeder Reactor Fuel Element (SUPERPHENIX)

Boron carbide absorbs the neutrons and thus allows to control the reactor power by moving these absorber elements axially into or out of the core.

The primary sodium is pumped upwards through the fuel elements and heated from 380 to 550 °C.

The loading and unloading of fuel, radial blanket and absorber elements is done with special refueling machines from the top of the reactor vessel. All LMFBR, therefore, have a rotating plug system at the top of the reactor vessel. This rotating shield plug system consists of an eccentric arrangement of up to three smaller rotating plugs which can be moved against each other. By means of the rotating plug system all positions of the different fuel elements of the honeycomb structure core can be reached (see Figure III-8). The sodium level within the reactor vessel can be lowered or increased by changing the pressure of the covergas argon which separates the upper hot sodium surface from the colder structures of the rotating shield plug.

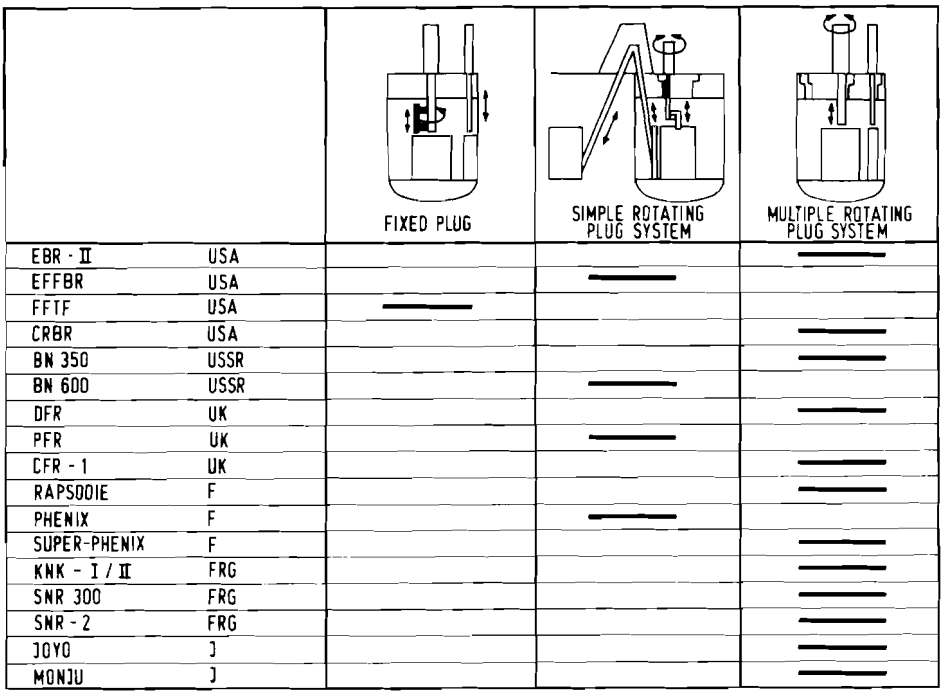


Figure III-8: Fuel Element Handling Systems for Fast-Breeder Reactors

Similarly as with the fast reactor core, much care is required for the design and construction of sodium-water steam generators. LMFBR steam generators contain non-radioactive secondary-circuit sodium and water, both separated from each other only by the tube walls. Because of their different physical and chemical properties, sodium and water lead to a considerable chemical reaction if they get into contact. Many design aspects have to be taken into account, such as fabrication, safety, operational availability, leak detection, inspectability, corrosion effects, repair, etc. Therefore, much research within the LMFBR programs has been devoted to steam-generator development and testing in 1:1 scale test facilities. Figure III-9 exemplifies two different concepts of steam generators, a straight tube steam generator and a helical tube steam generator of the SNR 300 plant. A number of other design concepts are used by other fast reactor projects [III-104]. Sodium-water reactions occur if one of the water-filled tubes in a steam generator fails. Water or steam is then released under high pressure into the sodium. Sodium-water reaction, hydrogen production and pressure generation result. A special pressure relief system using rupture discs then releases hydrogen to the atmosphere [III-105]. Since the secondary-circuit sodium does not become radioactive, no radio-activity is released to the atmosphere during a sodium-water reaction in a steam generator.

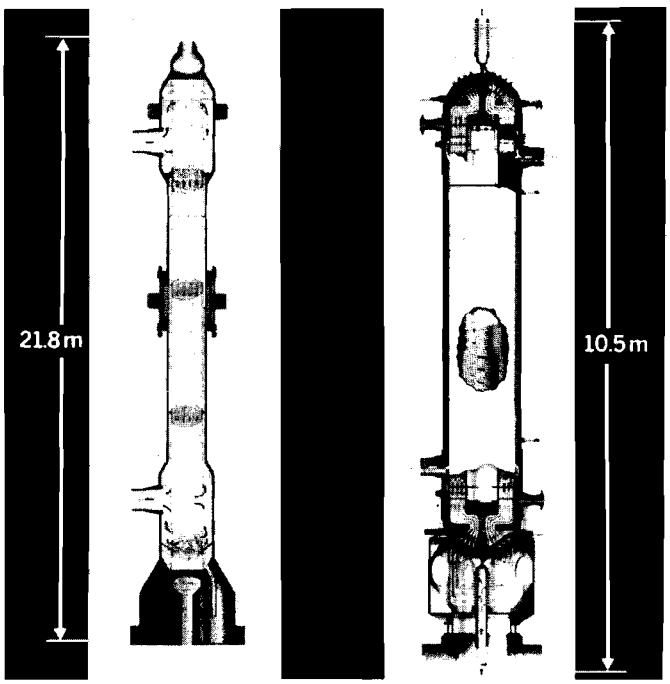


Figure III-9: Steam Generators of SNR 300 Fast-Breeder Prototype Power Plant

Much research has been carried out in the past ten years investigating these sodium-water reactions and the design optimization of pressure relief systems. There is no doubt that large LMFBR steam generators can be built upon a sound and safe technological concept. The analysis of the difficulties with steam generators in the early operation phase of BN 350 and PFR showed that there would be no major technological problems with LMFBR steam generators. However, much care must be devoted to the right choice of tube steel (intercrystalline corrosion, weldability, etc.) and to quality assurance during fabrication, if the expected high plant availability of future large commercial size LMFBR is to be reached.

1.3.2 State of Development of Sodium-Cooled Fast Breeder Reactors

With the construction and operation of the experimental reactors EBR-II in the USA, DFR in the UK, RAPSODIE in France, and BOR 60 in the USSR, it was already proven that it is possible to satisfactorily operate fast reactors with sodium as coolant over long periods of time. Their operation, including maintenance and handling of fuel and coolant, has become a matter of routine now [III-106 to III-109].

In addition, one must consider the broad experience which was obtained from numerous large scale out-of-pile sodium component test facilities. Such out-of-pile test facilities have been built and operated in the USA, UK, France, The Netherlands, FRG, and Japan. These sodium component test facilities allowed to gain experience with big components (pumps, heat exchangers, steam generators) in advance to their later use in the fast reactors of the 300 MW(e) class.

Design and construction of fast breeder reactors of the 300 MW(e) class began between 1965 and 1970. At that time the size of 300 MW(e) was considered a first step towards the construction of large commercial-size breeders of the 1000 to 1500 MW(e) class. Some of the 300 MW(e) class LMFBR are already in operation [III-110 to III-113], as can be seen from Table III-II and Figure III-10.

The BN 350 built at Shevshenko on the Caspian Sea in the USSR became critical in November 1972. After some difficulties with steam generators it now operates at 60% full power. Besides electricity generation, it is also used for sea-water desalination. The second Russian prototype fast breeder reactor BN 600 with 600 MW(e) output is still under construction at Bjelojarsk; its start of operation is scheduled for 1977 [III-114].

The French prototype fast-breeder reactor, PHENIX, reached its first criticality in August 1973, and since August 1974 has been operating perfectly at full power with an overall availability

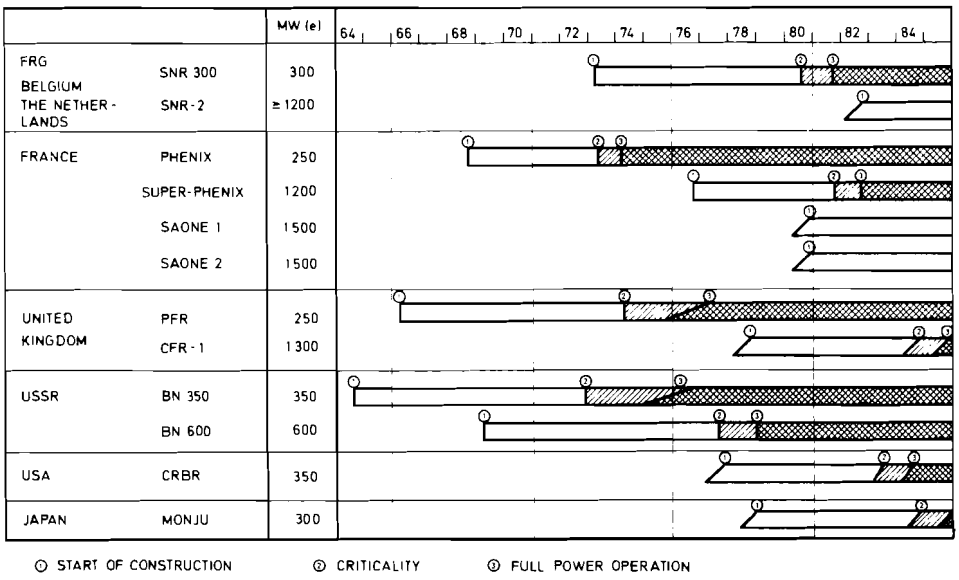


Figure III-10: Time Schedules for Different International Fast Breeder Projects

of 84%.* The mixed oxide fuel elements of PHENIX reached their maximum design burn-up of 70,000 MW(th)·day/t early in 1976 without any fuel pin failures [III-115].

The British prototype fast-breeder reactor PFR of 250 MW(e) became critical in March 1974. During power ascension leaks were detected in two of the three steam generators. Due to these difficulties it was operating at only 30% full power during 1975. Since the fall of 1976 PFR has been operating at 82% full power.

The 300 MW(e) German/Belgian/Dutch prototype reactor SNR 300 at Kalkar, FRG, is still under construction [III-116]. Criticality is planned for 1980. The start of construction of two other prototype fast reactors, the 350 MW(e) US demonstration plant CRBR at Clinch River in Tennessee, USA [III-37], and the 300 MW(e) Japanese prototype fast reactor MONJU [III-38], is scheduled for 1977 and 1978, respectively. In the US a very broad fast reactor research and development program is pursued; within this program it was decided to conduct the fast flux test facility FFTF

* PHENIX was shut down at the beginning of October 1976 due to a sodium leak in an intermediate heat exchanger, which is currently being repaired.

for materials research before the fast power reactor demonstration plant CRBR. FFTF is scheduled to go into operation in 1978 [III-117].

It is quite natural that during the construction and early operation phases of these prototype reactors certain technical difficulties (e.g. steam generator leaks) have to be overcome in some kind of learning process. So far there have been no basic problems with the construction of such LMFBR plants. The further successful operational availability of this new technology will, however, depend upon the careful plant design which in some cases must be prepared by preceding R&D work and with good quality assurance during construction.

1.4 Commercial Feasibility

The next step towards commercial feasibility will be the construction of large commercial-size fast-breeder reactors of an electrical output in the range of 1200 to 1500 MW(e). The Russian BN 600 is a first step in this direction. In France the start of construction of the 1200 MW(e) plant SUPERPHENIX is scheduled for 1976. Preparations on its future site at Creys-Malville by the Rhone River have started [III-118]. The plant will be operated by the Franco-German-Italian utility group NERSA. Two additional large-size (1600 to 1800 MW(e)) LMFBR plants in France are already being discussed by EdF; they will follow SUPERPHENIX after 1980. In the UK the CFR-1, a 1300 MW(e) LMFBR plant, has been under active design for a few years [III-46]. Having been postponed several times, it is now planned for around 1978.

Design work for similar large-size plants in the range of 1200 to 1600 MW(e) is also done by the German/Belgian/Dutch project [III-119] and in the USSR [III-48]. If all these projects are realized in time, the first commercial-size LMFBRs, SUPERPHENIX and CFR-1, will go into operation around 1982 to 1985.

The future LMFBR development will show whether these ambitious plans and construction schedules are feasible. The success and the early commercialization of the LMFBR will depend very much upon the design and the construction quality of big sodium components (pumps, heat exchangers, steam generators), improved core structural material to overcome irradiation damages at high fuel burn-up, no future delays in licensing such technically new large size plants, and their future good availability during operation. Commercial feasibility of LMFBR also includes the remaining part of the fuel cycle: fuel fabrication plants must perform under commercial conditions; fuel reprocessing plants and their waste disposal must be fully developed. Fortunately, the LMFBR system may again profit from LWR technology, which is about to enter a commercial fuel cycle within the next 10 to 15 years.

2. FUSION REACTORS

2.1 Scientific Feasibility

The prospects for demonstrating the scientific feasibility of man-made fusion power by the early 1980s are considered quite good. However, we must be careful as to how scientific feasibility is defined because it is a considerable step from that point to engineering feasibility. The criterion for this milestone depends on the method by which the thermonuclear burn is achieved, either by magnetic confinement of a low density hot plasma, or by the inertial confinement of a high density hot plasma inside an imploded pellet.

In the case of magnetic confinement, we can use the Lawson criterion [III-120]. The appropriate energy balance is given below for the D-T reaction, although it could be applied to other fuel cycles with proper modifications:

Electrical Energy Out = Electrical Energy In

$$\frac{n_D n_T}{2} \cdot \langle\overline{\sigma\, v}\rangle \cdot \tau \cdot w_{DT} \cdot \eta = \frac{3}{2} \cdot (n_D + n_T) \cdot k \cdot T_i \quad , \qquad (1)$$

where:

n_D, n_T are the deuterium and tritium fuel-ion densities, respectively ;

$\langle\overline{\sigma\, v}\rangle$ is the special averaged reaction cross-section for D and T at a fuel-ion temperature T_i ;

τ is the energy-confinement time ;

W_{DT} is the energy released per D-T reaction ;

η is the efficiency with which the thermal energy is converted to electrical energy ;

k is the Boltzmann constant.

If we assume a 50:50 mixture of D and T, a 40 per cent conversion efficiency, 22.2 MeV of energy released per reaction, and operation at the maximum $\frac{\langle\overline{\sigma\, v}\rangle}{T_i}$, we find that the product of the fuel-ion density and confinement time must be approximately

10^{14} sec/cm^3 at approximately 10keV. The corresponding numbers for the D-D reaction are roughly 5 · 10^{14} sec/cm^3 at roughly 100 keV, and roughly 10^{15} sec/cm^3 for the D-He3 reaction at roughly 100 keV.

The major thrust of research in the thermonuclear field has been to achieve the appropriate $n\tau$ values and ion temperatures simultaneously. A measure of the progress and state-of-the-art in this respect is shown in Figure III-11, where the achieved and anticipated values of $n\tau$ vs. T_i are given for the three basic magnetic confinement approaches: TOKAMAK [III-121], Mirror [III-122], and Theta Pinch [III-123]. Each approach has its advantages and disadvantages; e.g. Mirrors (2X, 2XIIB, and MX) can readily attain high temperatures but have a rather short confinement time because of high particle-leakage rates; Theta Pinches can increase the $n\tau$ value by operating at high fuel-ion densities but have achieved only lower temperatures up to now; and TOKAMAKs have achieved the highest $n\tau$ products but have difficulty in doing so at high plasma temperatures. However, the scaling laws seem to be fairly well understood for TOKAMAKs (T3, ST, ORMAK, TFR, and ALCATOR);

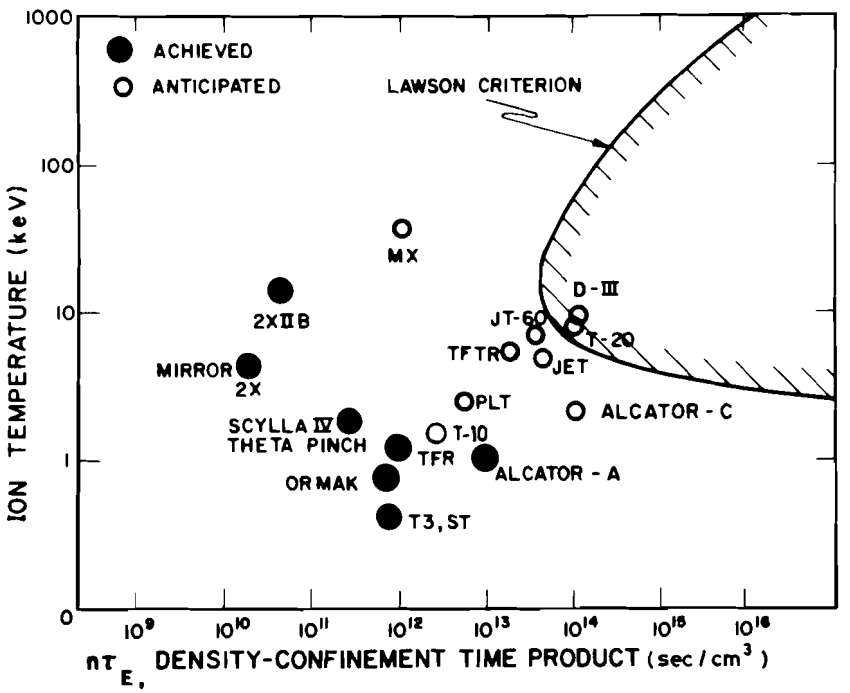


Figure III-11: "Break-even" Plasma Conditions for D-T Fusion Power

several larger experiments are planned in Europe, the Soviet Union, Japan, and the United States to demonstrate a scientific break-even with that concept. Two such experiments began operation in 1975 (T-10 [III-124], and PLT [III-125]), to produce plasmas in excess of $n\tau = 10^{12}$ sec/cm^3 with T_i being approximately equal to 1 to 3 keV. Table III-VII summarizes the rate of progress from 1955 to the present; it gives an idea as to how these achievements measure up to what will be needed for a commercial reactor.

Table III-VII: Plasma Parameters in Toroidal Devices[a]

Year	τ_E (sec)	T_i (Kelvin)	$n\tau_E$ (sec/cm^3)	Sustainment Time (sec)
1955	10^{-5}	10^5	10^9	10^{-4}
1960	10^{-4}	10^6	10^{10}	$3 \cdot 10^{-3}$
1965	$2 \cdot 10^{-3}$	10^6	10^{11}	$2 \cdot 10^{-2}$
1970	10^{-2}	$5 \cdot 10^6$	$5 \cdot 10^{11}$	10^{-1}
1976	$5 \cdot 10^{-2}$ (T-10, PLT)	$2 \cdot 10^7$ (TFR, ORMAK)	10^{13} (ALCATOR)	10^0 (T-10, PLT)
Needed for a Reactor	10^0	10^8	10^{14}	$\gtrsim 10$

[a] B. Pease, Culham Laboratory, UK.

Beyond the present machines, five large TOKAMAKs are being designed, all of which should come close to, if not exceed, the Lawson criterion. Two of these devices will use hydrogen plasmas (JT-60 [III-126], and DOUBLET-III [III-127]), while the other three will be the first D-T burning systems (T-20 [III-128], JET [III-129], and TFTR [III-130]). All of these machines should be in operation by the late 1970s or early 1980s. It should be noted that the object of the TFTR is to demonstrate a limited form of energy break-even by injecting high energy D and T into a plasma to induce fusion reactions in addition to those from the ion-ion thermal reactions. This concept is called a two-component plasma and is explained elsewhere [III-130]. The effective break-even conditions for $n\tau$ in this concept are lower than those for an ignited system, but it is

questionable whether such a system could be a commercial reactor.

In summary, the prospects for achieving a meaningful scientific demonstration of D-T fusion within the next five to seven years are quite good for TOKAMAK reactors. The situation is not so clear for the Mirror and Theta Pinch concepts, although work will still continue in those areas. Because of the optimism in the TOKAMAK area and since most of the advanced reactor design studies have been done in that area, we will narrow our scope of consideration to that concept for the rest of this report. This is only done because of the limited extent of this subsection; future breakthroughs in the other systems may make them more desirable than the TOKAMAK.

A similar treatment of scientific feasibility for laser fusion must take a slightly different approach. It is presently envisioned that pellets containing D-T would be imploded to high densities approaching $\rho R \cong 3$ g/cm^2, where R is the radius of the imploded fuel, and ρ its density. For reference purposes the density of solid D-T at one atmosphere is approximately 0.2 g/cm^3. The burning process is initiated when a small central portion of the fuel pellet is compressed to roughly 0.3 g/cm^3 raising the temperature to approximately 5 keV, thus igniting the fuel. The alpha particles deposit their heat in this region, and the burn front will propagate outward, causing about 30 per cent of the fuel to be "burned" before disassembly. This would give a yield of 10^{11} J/g. Such a reaction yields the following energy balance [III-131]:

Electrical Energy Out = Electrical Energy In

$$(G_o + 1) \cdot E_L \cdot f_R \cdot \eta = \frac{f_H}{E_I} \qquad (2)$$

where:

G_o = gain of the fuel

$$= \frac{\text{thermonuclear reaction energy released}}{\text{internal energy of all the fuel if heated to ignition}} \ ;$$

E_L = efficiency of laser

$$= \frac{\text{laser energy out}}{\text{electrical energy in}} \ ;$$

f_R = fraction of electrical energy used to pump the laser

η = conversion efficiency from thermal to electrical energy;

$$f_H = \frac{\text{internal energy of imploded pellet}}{\text{energy required to heat imploded pellet to a uniform temperature of 5 keV}}$$

E_I = efficiency of implosion

$$= \frac{\text{internal energy of imploded fuel}}{\text{laser energy in}} .$$

Scientific feasibility could be defined as when the gain of the pellet

$$G_p = \frac{(G_o + 1) \cdot E_I}{f_H}$$

$$= \frac{\text{thermonuclear yield from pellet}}{\text{input energy from the laser}} ,$$

reaches a value of one. However, a more meaningful definition of scientific feasibility might be the point where all of the output energy is fed back into the lasers ($f_R = 1$), and a thermal conversion efficiency of approximately 40 per cent is assumed. The relationship between G_p and E_L is then given in Figure III-12 [III-131], along with an estimate of the efficiencies of glass and gas lasers. In the most optimistic case of a laser efficiency of ten per cent we see that a minimum value of G_p required would be 25. The required pellet gain would rise to 500 to 1000 if glass lasers were developed to their estimated maximum-efficiency range. From these considerations it is clear why gas lasers have potential reactor applications, while glass lasers probably do not. However, there is a problem with the coupling of CO_2 laser light (10.6 micron) to pellets, and it is possible that inordinately high powered lasers will be required.

No simple picture can be given as to the state-of-the-art for laser fusion, but several recent reviews should be helpful to the reader [III-131 to III-135]. Laser-induced implosions have already been demonstrated [III-134, and III-135] with 10^3 J lasers; the next step is to increase the laser energy to 10^4 J to achieve pellet gains of approximately 0.1. Workers at the Lawrence Livermore Laboratory in the US project that this may

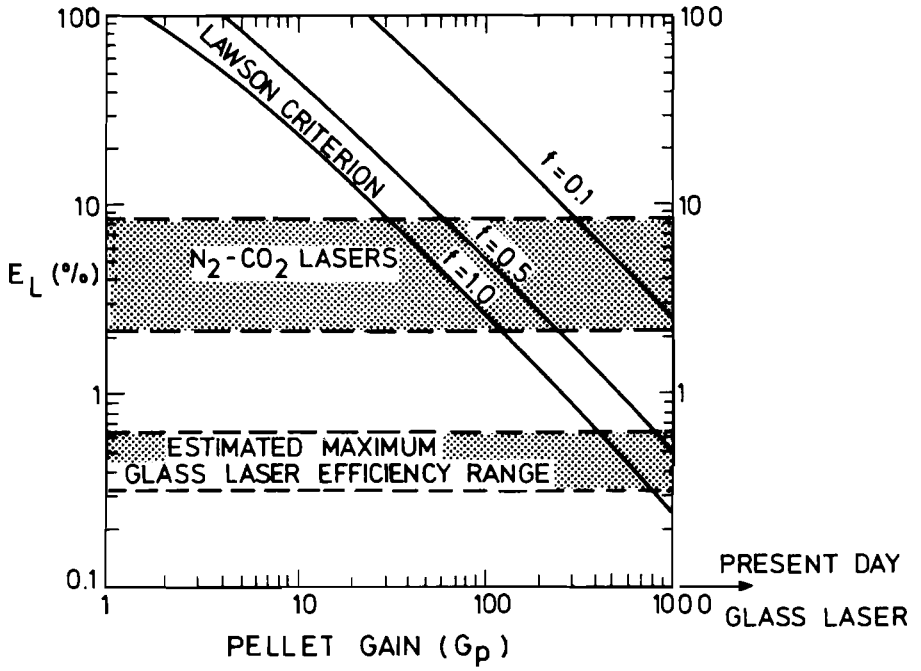


Figure III-12: Laser Efficiency and Yield Ratio Requirements for Inertially Confined Fusion Reactors

occur in the 1978 to 1980 period through the construction of several progressively larger lasers, as shown in Figure III-13. The details of the plan can be found elsewhere [III-136]. Pellet gains of approximately 25 requiring 100 kJ laser systems are not anticipated to be achieved until 1981 to 1983.

In summary, the laser fusion program is clearly limited at this time by high-energy (100 kJ), short-pulsed (<1 nsec), high-efficiency (5% to 10%) lasers. The rate of progress is significant, and projections are that scientific feasibility demonstrations should take place at about the same time as those for magnetic confinement, i.e., in the early 1980s.

2.2 Engineering Feasibility

In order to progress from scientific to engineering feasibility one must demonstrate that the release of fusion energy can generate *net* power on a reasonably reliable basis over *long* periods of time. This step will be a large one, perhaps as large as the demonstration of scientific feasibility itself. It is beyond the scope of this discussion to outline all of the

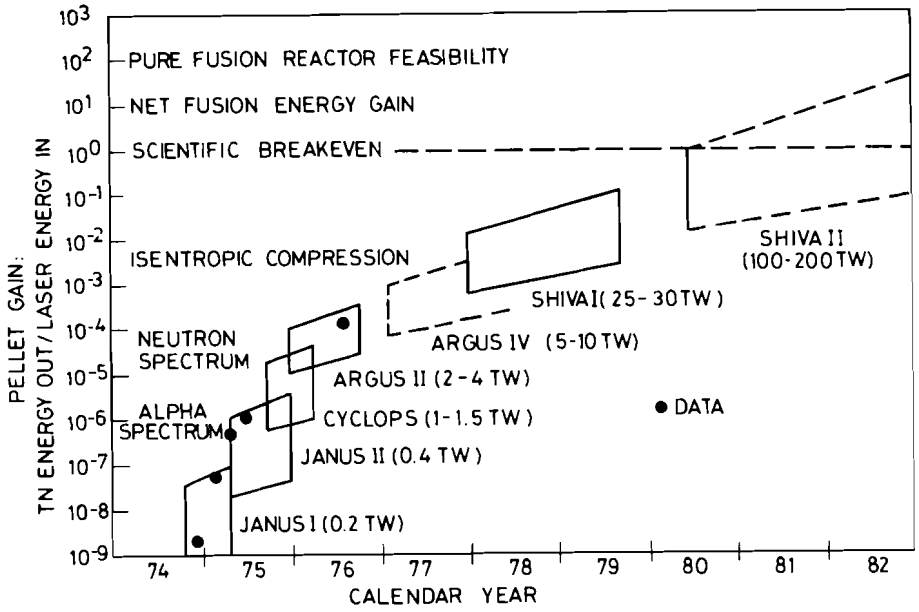


Figure III-13: Laser Fusion Energy Yield Projections

steps and technologies that must be mastered for each confinement approach. That has been the subject of many conferences [III-137 to III-140], workshops [III-141, and III-142], and documents [III-143 to III-146]. However, we can briefly summarize which major technologies are required, and give an indication of how such problems may be solved in the US fusion program.

First of all, there are some problems which are common to all types of D-T fusion reactors, such as radiation damage to the solid members of the reactor first wall, the attainment of breeding ratios greater than one using various types of lithium containing materials, and the handling of large amounts of highly volatile tritium. All of these areas are receiving major attention in the countries with fusion programs, and it appears that at least ten to twenty years may be required before these problems can be completely solved. Moving to the D-D cycle only partially alleviates the radiation damage problem as we will see later, but it would eliminate the breeding problem and significantly reduce the tritium handling problems. Secondly, there are some problems which are unique to particular reactor concepts, such as the development of large-bore superconducting magnets for the TOKAMAK and Mirror reactors.

Other examples include fast energy switching for Theta Pinches; repetitive fast and efficient lasers for laser fusion; heating and fueling devices for TOKAMAKs; direct convertors for Mirrors; windows and mirrors for laser fusion; load-leveling schemes to deliver power during the periods between burns of TOKAMAK; fatigue-resistant structural materials for laser and Theta Pinch reactors; and so forth. The solution to these problems will only come after extensive testing in specially designed facilities that will themselves require great ingenuity to design and build.

A detailed program plan to overcome both the plasma-physics and technology problems for the TOKAMAK has been recently described [III-147, III-148]. Although the majority of the planning has been done for the TOKAMAK scheme, other plans are available for the laser, Mirror and Theta Pinch approaches. We will only describe the TOKAMAK plan here.

The US approach has been formulated in terms of the level of effort that the governmental organizations are willing to find and that the scientific community is willing to support. These fall into five general categories as listed below (labeled as logics for this discussion):

Logic I: Level of research effort
Research and development are supported at an arbitrary level in order to develop basic understanding. (If this pace were continued, a practical fusion power system might never be built.)

Logic II: Moderately expanding, sequential
Funds are expanding but technical progress is limited by the availability of funds. Established commitments are given funding priority but new projects are not started until funds are available. In spite of limited funding a number of problems are addressed concurrently. (At this rate, a fusion demonstration reactor might operate in the early 21st century.)

Logic III: Aggressive
The levels of effort in physics and engineering are expanded according to programmatic need, assuming that adequate progress is evident. New projects are undertaken when they are scientifically justified. Many problems are addressed concurrently. Funding is ample but reasonably limited. (This program would be aimed at an operating demonstration reactor in the late 1990s.)

Logic IV: Accelerated
A great many problems are addressed in parallel, and new projects are started when their need is defined. Fabrication and construction are carried out on a normal basis with enough priority to minimize delays. The availability of funds is still limited but a secondary factor in program

planning and implementation. (This approach would be aimed at demonstration reactor operation in the early to mid-1990s.)

Logic V: Maximum effective effort
Manpower, facilities and funds are made available on a priority basis; all reasonable requests are honored immediately. Fabrication and construction are expedited on a priority basis so that completion times for major facilities are reduced to a practical minimum. (An operating demonstration plant around 1990 would be the program goal.)

While the major difference betweeen the logic sequences above may appear to be associated with costs and goal dates, the reader should recognize that the degree of risk varies greatly from one sequence to another. That is, there is no assurance that with an infinite amount of money the physics problem can be overcome soon. Nevertheless, such an exercise can be useful in establishing the most *optimistic* dates for certain achievements.

The final results of such an exercise are shown in Figure III-14 where the estimated yearly operational (not capital) budgets are plotted versus time. A locus of points for the projected operation of the demonstration reactor is given, although it is highly unlikely that the *logic sequence V* could ever be met (or even *Logic IV*). In the author's opinion *Logic III* is probably the most optimistic sequence that could be achieved, and we will pursue that in more detail below. Before we leave the general logic structure it should be noted that US ERDA estimates on the total cost (capital and operating) to the sucessful operation of a demonstration plant are as listed below:

Logic	$ 10^9 (1978 equivalent dollars)
I	undetermined
II	16.2
III	15.5
IV	14.8
V	20.1

Specifically addressing the *Logic III* sequence, we have depicted in Figure III-15 the timing of the plasma-physics and technology devices required for the demonstration power plant (DPR). It should be recognized that *Logic III* assumes that the currently planned plasma-physics devices, i.e. PLT, PDX, and D-III, will satisfy their design goals and feed into future systems. The TFTR device, which will be the first D-T burning device in the US, should continue to test current scaling laws with alpha-particle heating, demonstrate auxiliary heating techniques, and provide some information on fueling methods and

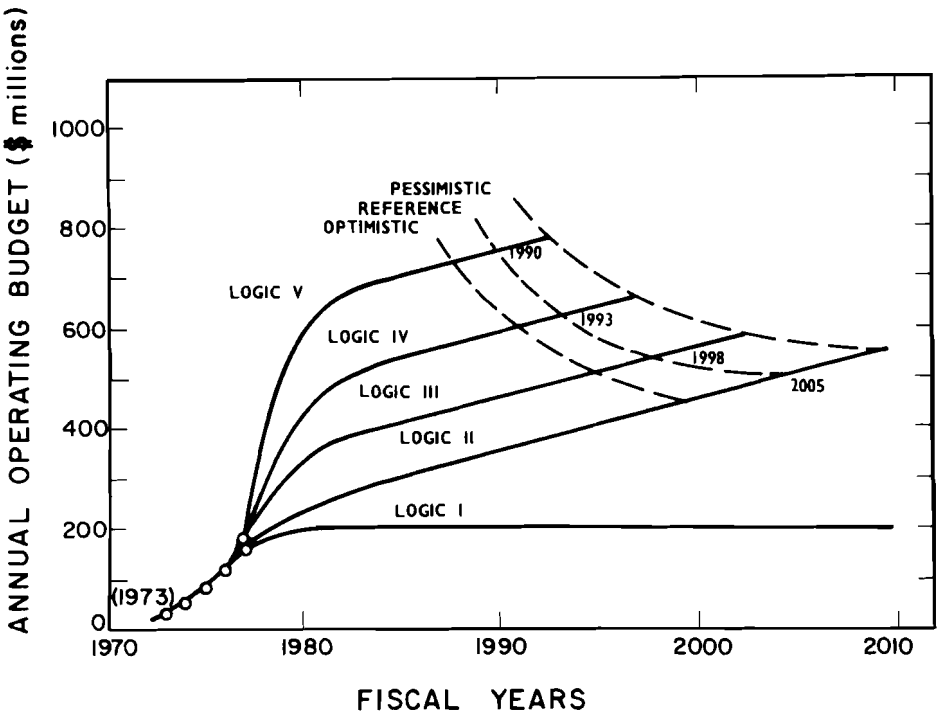


Figure III-14: Summary of US ERDA Projected Costs to a Fusion Demonstration Power Reactor via Different Logic Sequences

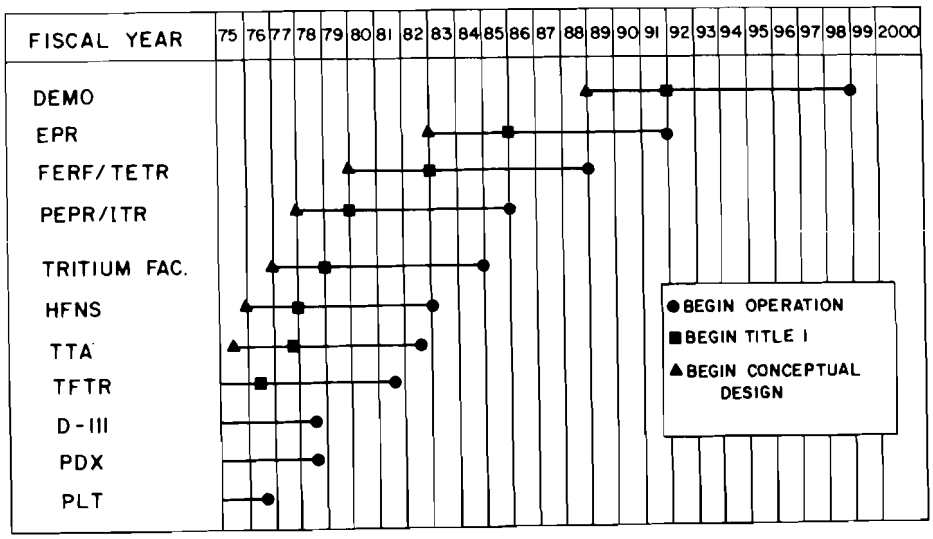


Figure III-15: Proposed US ERDA Program to Develop a TOKAMAK Demonstration Reactor

impurity control. Results from the TFTR will be coupled with the JET, T-20, and JT-60 reactors.

At the same time as the TFTR is being constructed, several auxiliary technology facilities are being designed for operation in the 1980 to 85 period. For example, construction of a large superconducting-magnet test assembly (TTA) will begin in fiscal year 1977, followed shortly be construction of a high-flux neutron facility (HFNS). A tritium handling facility is also currently being designed. All of these facilities are expected to assist in the design and construction of the EPR (experimental power reactor) of the 1991 time period.

Before the EPR, several other facilities have been postulated. There will be a need for one more step between the TFTR and the EPR in terms of plasma scaling and ignition physics, along with the development of associated plasma-physics technologies. Such devices are now labeled PEPR (prototype experimental power reactor) or ITR (ignition test reactor). The PEPR/ITR would be the first large-scale facility to make use of superconducting magnets. There is also a great need (as we shall see in Chapter IX) for a large-volume materials test facility. Two such proposals are currently being considered; a FERF (fusion engineering reactor facility, [III-149]), and TETR (TOKAMAK engineering test reactor, [III-150]). These devices would provide one to two m^3 of high flux (≈ 1 MW/m^2) test volume to screen and qualify materials for the DPR (demonstration power reactor), and to a limited degree act as a verification of materials already chosen for the EPR.

The EPR, anticipated to operate in 1991, will produce electricity in the several hundred MW(e) range thereby demonstrating an integrated power cycle, and should also demonstrate an integrated tritium breeding system, at least in a few modules. This will be the first power reactor which experiences significant radiation damage, and as such will provide valuable reliability experience for the DPR.

The DPR, as envisioned today, would be the first fusion device to provide significant net (but not necessarily economic) electrical power and its own tritium (i.e. BR $\approx$ 1). Operating plant factors of approximately 70 per cent will be a prime goal of this reactor. All the components necessary for commercialization will be present in this device, and future commercial reactors will be more optimized designs of this reactor, but not radically different.

As was noted earlier, *Logic III* may be unattainable if even small physics and technology difficulties arise. Others [III-151 to III-153] have proposed a more extended and diverse program which would produce a DPR in the year 2005 to 2010. (The recent *Logic II* is close to this plan.)

Finally, it is worthwhile to note the funding level of US magnetic and inertial confinement programs. Detailed values are listed in Table III-VIII and graphically depicted in Figure III-16. The total US fusion program from 1951 to 1977 amounts to 1.8307 billion (actual) dollars, or roughly 2.2 billion dollars adjusted to 1975 equivalent. To the US ERDA figures one should add approximately 20 million dollars from utilities, 50 million dollars from industry, and another 30 million from universities, state, and other federal agencies. These figures reveal that in the US alone, 1.9 billion dollars will have been spent on fusion through fiscal year 1978. Of this amount, roughly 75 per cent (roughly 1.4 billion dollars) has been spent for the magnetic approach, and 25 per cent (roughly 500 million dollars) for the inertial-confinement scheme. Future costs of the US program appear to be on the 15 to 20 billion dollar level for the facilities outlined in Figure III-15. Since the US effort was only roughly one-sixth of the world effort in 1973 (see Table III-IX) and probably will be roughly one-third in 1977, one might expect that roughly 50 billion dollars could be required, world wide, to bring fusion to the point of engineering feasibility by the year 2000.

2.3 Commercial Feasibility

The ultimate objective of fusion research today must be aimed at commercial feasibility, otherwise the world would not invest the previously estimated large amount of resources for CTR development. Commercial feasibility can be defined as the point at which public or private utility organizations would purchase a fusion reactor on the open market in free competition with hydro-, fossil and fission power stations. It is entirely possible to demonstrate that fusion reactors are feasible from an engineering standpoint but then to discover that they cannot compete on the open market on account of inherently high costs because of their low power densities. This means that the capital costs of fusion power plants are likely to be as high or higher than for current fission or fossil-fueled power plants. However, as fuel costs rise for the competing sources of energy or as problems develop with the environmental aspects of the other energy sources, it is possible that the commercial feasibility of fusion power could occur early in the 21st century. Once commercial feasibility is achieved, it is quite likely that it will take another twenty to thirty years before fusion could provide a significant amount (more than 20 per cent) of the world's electrical energy needs [III-155].

Table III-VIII: Historical Funding Level for the US Fusion Program (million dollars)

Fiscal Year	Magnetic Confinement			Inertial Confinement			Grand Total All Fusion
	Operating	Construction and Capital	Total	Operating	Construction and Capital	Total	
1951-53	1.1		1.1				1.1
1954	1.8		1.8				1.8
1955	4.7	1.4	6.1				6.1
1956	6.6	0.8	7.4				7.2
1957	10.7	0.9	11.6				11.6
1958	18.4	10.8	29.2				29.2
1959	27.0	1.9	28.9				28.9
1960	31.0	2.7	33.7				33.7
1961	29.0	1.0	30.0				30.0
1962	23.6	1.2	24.8				24.8
1963	24.2	1.3	25.5	0.2		0.2	25.7
1964	21.0	1.6	22.6	1.1		1.1	23.7
1965	21.3	1.8	23.1	1.3		1.3	24.4
1966	21.8	1.3	23.1	1.2		1.2	24.3
1967	22.4	1.5	23.9	1.4		1.4	25.3
1968	24.7	1.9	26.6	1.3		1.3	27.4
1969	26.5	3.2	29.7	2.1		2.1	31.8
1970	27.7	6.6	34.3	3.2		3.2	37.5
1971	28.3	3.9	32.2	9.0	0.4	9.4	41.6
1972	31.0	2.3	33.3	15.9	3.5	19.4	52.7
1973	37.0	2.2	39.1	23.4	11.1	34.5	73.6
1974	53.0	3.3	56.2	34.3	10.0	44.3	100.5
1975	88.9	6.4	95.3	43.4	20.3	63.7	159.0
1976	115.3	35.0	138.9	59.5	20.0	79.5	218.4
1976[a]	41.7	10.1	47.6	18.0	2.4	20.4	68.0
1977[b]	176.1	123.5	230.2	69.3	29.6	98.9	329.1
1978[b]	196.6	75.8	272.4	?	?	120.1	392.5
Total	1111.4	217.3	1328.7	284.6 +(1978)	97.3 +(1978)	502.0	1830.7

[a] Transition

[b] Projection

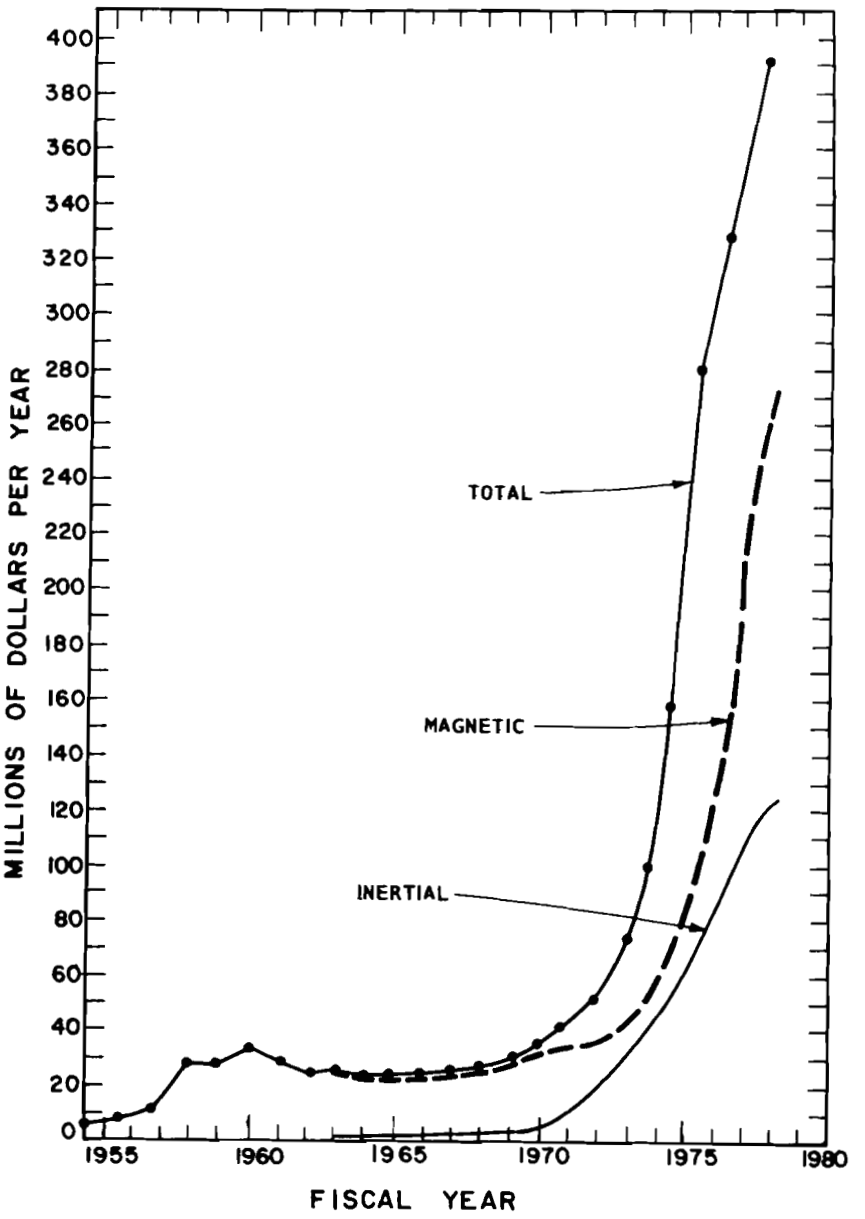


Figure III-16: Fusion Funding in the United States

Table III-IX: Summary of World Efforts in Fusion Research, 1973 [III-154]

Country	% of World Effort
USSR	37.1
USA	15.8
FRG	13.0
Japan	9.3
France	7.2
UK	4.5
Others	13.1
Total	100.0

2.4 Scientific Feasibility of Laser Fusion

(R.R. Grigoriants)

The idea of using lasers to heat up a plasma to temperatures that allow for fusion was first conceived in the early sixties [III-156].

Scientific feasibility of laser fusion is mainly a question of heating and compressing the thermonuclear fuel in the pellet.

Similarly to the way in which the Lawson criterion is deduced, one may deduce the minimal laser energy E_{las} [III-157] to get a certain value $K_{BR}{}^{t}$ (E_{las} is measured in MJ):

$$E_{las} = 10 \cdot (K_{BR}{}^{t})^{3} \cdot \eta^{-4} \cdot \left(\frac{n_s}{n_o}\right)^{2} \qquad (3)$$

where

$K_{BR}{}^{t}$ = ratio of energy released by fusion and that of laser radiation supplied to the pellet;

η = part of laser radiation energy transformed into heat energy of the plasma;

n_s = density of particles of solid hydrogen;

n_o = density of particles of deuterium and tritium during compression of target.

Equation (3) takes into account that the temperature of the plasma is raised owing to the release of fusion energy; however, on account of a number of other assumptions it is only an approximation. From this equation it can easily be seen that in order to get $K_{BR}t = 100$ at $E_{las} = 10^5$ J it is necessary to reach a high density (about 10^3 n_s), which, in turn, requires a high pressure up to 10^{11} atm at fusion temperatures of more than 10^7 K.

The fundamental solution to this question consists in the use of the reactive pulse that is generated when the absorption of the laser radiation causes the outer layer of the target to burst.

The entire process of initiating laser fusion and burning of the pellet can be divided into three stages:

- The outer part of the pellet evaporates and moves towards laser radiation, thus forming a corona;
- There is a movement of matter towards the center of the pellet-target; the inner part of the target is heated up to fusion temperatures and is compressed up to a density of 10^{21} to $10^{26}/cm^3$;
- Fusion starts in the center of the target, resulting in a release of energy. Part of this energy is used for heating the pellet, which speeds up the further release of energy. So the pellet is heated even more and finally explodes.

The entire process takes about 10^{-9} sec; intensive burning lasts for approximately 10^{-11} sec.

For an optimum burning of the target it is necessary to control the timing of the heating of the individual layers of the pellet. This can be done either by profiling the laser pulse or via the building of the target shell. The first method complicates the laser system, the second calls for a technology which would make it possible to have individual pellet layers of a thickness of a few or dozens of microns.

Numerous present experiments on the heating of fusion pellets show that it is possible to obtain energy ratios ($K_{BR}t$) of as high as 100 to 1000.

The basic problems of physics connected with the creation of a laser fusion reactor (LFR) are the following:

- the problem of creating radiation in big and sufficiently efficient laser systems;

- the problem of reciprocal action between laser radiation and target;
- the creation of methods to diagnose the processes taking place in the plasma.

The requirements for the capacity of laser systems are illustrated in Table III-X.

Table III-X: Results of Calculations of Microexplosions of Laser Fusion Targets [III-158 to III-161]

Laser Energy E_{las} (J)	Target Mass (g)	Breeding Coefficient E_f/E_{las}	Energy of Microexplosion E_f (J)	Total Number of Neutrons
10^4	$6 \cdot 10^{-6}$	80	$8 \cdot 10^5$	$2.8 \cdot 10^{17}$
10^5	$6 \cdot 10^{-5}$	110	$1.1 \cdot 10^7$	$3.9 \cdot 10^{18}$
10^6	$1.5 \cdot 10^{-2}$	1000	10^9	$3.5 \cdot 10^{20}$

The energy balance of a LFR plant clearly illustrates the necessity for a sufficiently high laser efficiency.

The energy-block efficiency with laser but without laser heat is:

$$\eta' = \frac{K_{BR} \cdot \eta_{CS} - \frac{1}{\eta las}}{K_{BR} - 1} \quad , \tag{4}$$

where:

η_{CS} = conversion system efficiency (determined, for example, by the specific heat consumption of a turbine);

K_{BR} = ratio of fusion energy and of laser radiation energy supplied to the pellet; K_{BR} also allows for energy breeding in the blanket (for example, in the nuclear reaction Li6 + n → T + He, or in the fusion blanket of a hybrid reactor);

η_{las} = ratio of laser radiation supplied to the pellet and of energy supplied to the laser system; the latter includes the energy supplied to auxiliary units (for example, to the compressor pumping the $N_2 + CO_2$ mixture, or to the electrolyzer in the HF-laser).

The difference between the conversion system efficiency and the efficiency of the laser driver block may be expressed by:

$$\eta_{CS} - \eta' = \frac{\frac{1}{\eta_{las}} - \eta_{CS}}{K_{BR} - 1} \quad . \tag{5}$$

It is obvious that if

$$\eta_{las} < 0.1 \quad ,$$

the conversion system efficiency affects the $(\eta_{CS} - \eta')$ value only slightly.

For the full or partial utilization of laser energy one obtains the following expression [III-162]

$$\eta_{\nu} = \eta' + \frac{\nu\left(\frac{1}{\eta_{las}} - 1\right)}{K_{BR} - 1} \eta_{CS} \quad , \tag{6}$$

where ν is that part of energy which is supplied to a laser but not converted into light energy.

For an experiment for demonstration purposes in which energy is not released into the outer network, with

$$K_{BR} = 100$$

and

$$\eta_{CS} = 0.45$$

it is sufficient to have with full utilization of laser heat

$$\eta_{las} = 0.032 \quad ;$$

and without utilization of laser heat

$$\eta_{las} = 0.022 \quad .$$

Although the calculations according to (6) are rough approximations (e.g. they do not take into account heat loss in pipes, etc.), they do allow one to estimate the correlation between K_{BR} and η_{las}, and the influence of laser heat utilization.

For laser fusion, several types of lasers have been suggested:

- neodym glass laser;
- CO_2 laser of high pressure;
- iodine laser;
- chemical hydrogen fluorine laser.

It can be expected that the efficiency of the neodym laser amounts to 1 to 2%, and that of the CO_2 laser to 5 to 10%. The efficiency of the chemical laser will essentially be determined by the efficiency of the electrolyte.

The neodym laser is the laser most thoroughly developed today. Especially designed for spherically symmetrical target-irradiation, the nine-duct neodym laser was introduced in the USSR in 1970. Radiation energy changed from 200 to 1300 J, and pulse duration from 1 to 16 nsec [III-163]. Similar plants have been created in the US. The success in developing lasers and the realization of symmetrical irradiation of pellets made it possible to proceed from numerical to practical experience (see Table III-XI).

At present a unanimous explanation of the reasons for neutron emergence is not available.

The construction of big new laser plants will lead to the next step in experimental research of laser plasma. The USSR is developing a neodym glass laser with an energy of approximately 15 kJ, a pulse duration of 1 to 20 sec, and η_{las} equaling 0.01 to 0.02 [III-168]. The development program of laser fusion engineering in the US is described in section III.2.2.

In the opinion of leading Soviet scientists [III-157, and III-158] experimental and theoretical research has shown that it is possible to solve many basic problems of fusion physics. One may, with sufficient conviction, speak of the existence of an efficient absorption of laser radiation, and of a considerable compression of targets.

Table III-XI: Results of Experiments in Laser Heating and Compression of Deuterium and Deuterium/Tritium Targets.

	[III-164]	[III-165, III-166]	[III-167]
Laser Energy J	200	30-60	20
Pulse Duration nsec	6	0.3	0.1
Maximum Radiation Flux W/cm^2	$3 \cdot 10^{14}$	$5 \cdot 10^{15}$	-
Amount of Absorbed Energy %	70-80	5-10	60
State and Size of Target μ	solid CO_2 R ≈ 25-100	totally glass with D + T	fully spherical R ≈ 100
Registration Method of Compression	for neutrons D-T reaction	camera obscura	camera obscura
Compression, Achieved Density g/cm^3	30; 30	125; 0.5	-
Temperature of Thermonuclear Fuel keV	0.5	0.7	0.1-2.3
Neutron Output	$5 \cdot 10^6$	10^5-10^6	-

Since the time when the idea of laser fusion was first conceived hardly more than a decade has passed, and the results of the research work carried out in that period have been really impressive.

3. CONCLUSIONS

The development and evolution of new major technologies seem to follow a pattern that distinguishes three thresholds of feasibility:

(a) scientific feasibility;

(b) engineering feasibility;

(c) commercial feasibility.

This pattern is important for judgments or assessments.

The development of fast breeder reactors first followed the line of metallic fuel elements and power-reactor sizes of only few hundred megawatts. It is represented by early concepts such as that of the Enrico Fermi Fast Breeder Reactor (EFFBR). The second line of fast breeder reactor development is distinctly different in terms of fuel technology, reactor physics, safety, as well as power plant characteristics; it uses mixed plutonium/uranium oxide as reactor fuel. After the BN 350 became critical in November 1972, it was the French PHENIX reactor which was the first reactor of the 300 MW(e) class that came to designed power (1974). It has been in full operation on the grid, with load factors in the neighborhood of 85%.* The British PFR and the Soviet BN 350 are also in operation now after initial engineering difficulties, mostly on the steam generator side, have been overcome. The German/Belgian/Dutch fast breeder prototype reactor SNR 300 is in the middle of its construction period; so is the Soviet BN 600, while construction of the US Clinch River Breeder Reactor (CRBR) and the Japanese MONJU reactor is expected to start soon.

The various fast reactor groups of the world now have available a large set of physics and engineering test facilities whose build-up was a major part of the overall effort, in terms of capital investment, manpower and time. The technologies for liquid sodium as a coolant and for mixed oxides as fuels are essentially in hand. A major share of out-of-pile and in-pile tests is devoted to proofing tests as required in the licensing process for large power reactors of the 1200 MW(e) class.

Preparations for the semi-commercial class of 1200 MW(e) are well under way in France, the UK, the FRG, and the US.

It can, therefore, be concluded that the thresholds of scientific and engineering feasibility of fast breeder reactors have been passed; the threshold of commercial feasibility, however, has not yet been passed. Present projections in the FRG and in France anticipate this threshold for about 1990.

For fusion power, demonstration of scientific feasibility means creating in an experimental device a combination of fuel density, temperature, and confinement time which would lead to a net output of energy in a reactor. No such scientific feasibility demonstration has yet taken place as of late 1976. Of the two main approaches to the problem--magnetic confinement and inertial confinement--magnetic confinement has the longer history (it originated in the early 1950s) and the greater number of variations (TOKAMAK, Mirror machines, and high-density

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* PHENIX was shut down at the beginning of October 1976, due to sodium leak in an intermediate heat exchanger which is currently being repaired.

pinches are the most important at present). The idea of inertial confinement for a fusion reactor dates from the early 1960s, and the two main variations are to use lasers or electron beams to initiate the required implosions. Many proponents of magnetic confinement believe that large TOKAMAK devices, now in the late stages of design or early stages of construction, will achieve scientific break-even (i.e. the Lawson criterion) in the early 1980s. (DOUBLET-III and TFTR, for example). Some advocates of laser fusion believe that such systems can also achieve scientific break-even by the early 1980s, although this view is more controversial and is clouded by classification of relevant results. Once scientific feasibility is achieved with either magnetic or inertial confinement, formidable problems of materials and engineering will have to be solved before technological feasibility can be demonstrated in the form of a working reactor. This is unlikely to be achieved before the year 2000. Commercial feasibility will not be assured even when such a reactor exists; it is possible that fusion will work but that it will simply be too expensive. Even if it does prove commercially feasible by the early twenty-first century, a contribution of as much as ten per cent of the electricity used in industrial nations still seems unlikely before the years 2020 to 2030.

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IV. REFERENCE REACTOR SYSTEMS

G. Kessler, G.L. Kulcinski

1. FISSION REACTORS

1.1 The Fast Breeder Prototype Reactor SNR 300

1.1.1 General Characteristics of the Plant

SNR 300 is a loop-type, liquid-metal-cooled fast-breeder reactor under construction by INB* at Kalkar, on the lower Rhine, FRG. The thermal power capacity of the plant will be 736 MW, its electrical power 312 MW. In the reactor core sodium is heated up from 377^0 to 546^{0}C. This heat is transferred to intermediate heat exchangers, located outside the reactor vessel, via three parallel primary loops. From the intermediate heat exchangers, three secondary (intermediate) sodium circuits transfer the heat to three steam generator systems, each of which is housed in a separate building. The steam conditions at the turbine inlet are 500^{0}C/160 bar.

1.1.2 The Reactor Core

The reactor core consists of cylindrical fuel zones containing UO_2 and PuO_2, which are surrounded by radial and axial blankets consisting of UO_2 with depleted U. The fresh core contains 1.3 t of plutonium. The Pu enrichment is approximately 25 and 35% in the inner and outer fuel zones, respectively.

Accordingly, the mean power density of nearly 300 kW/liter in the SNR 300 core is much higher than that in a light-water reactor. The diameter of the core is 1.78 m, its height is 0.95 m. The fast neutron flux (> 0.1 MeV) is up to $4 \cdot 10^{15}$ n/cm^2·sec,

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* INB (Internationale Natrium-Brutreaktor-Bau GmbH), is a joint subsidiary of the companies INTERATOM, BELGONUCLEAIRE, and NERATOOM. The plant will be operated by SBK (Schnell-Brüter-Kernkraftwerks GmbH), a joint subsidiary of the utilities RWE (FRG), SYNATOM (Belgium), and SEP (Netherlands).

the fast neutron fluence up to $1.4 \cdot 10^{23}$ n/cm^2 at the end of the lifetime. During reactor operation part of the uranium in the fuel zones and blankets is transformed into plutonium.

The core configuration of the SNR 300 is made up of hexagonal subassemblies (see Figure IV-1): 205 fuel and 96 blanket elements, 9 absorber elements in the control and first shut-down system, and 3 absorber elements in an independent second shut-down system. Each element has a width of 110 mm.

Each fuel element (see Figure IV-2) of the first core (Mark Ia) is made up of 169 rods spaced by grids. The outer diameter of such a fuel rod is 6 mm. Each rod consists of a thin-walled (0.38 mm) stainless-steel canning tube with fuel pellets and UO_2 pellets, making up the fuel zone and the upper and lower axial blankets. A fission gas plenum is at the lower end of each rod. The mean and maximum design fuel burn-ups are 55,000 and 87,000 MW(th)·day/t, respectively. For a commercial breeder-reactor power station, even higher burn-ups are envisaged.

Each radial blanket element contains 61 rods of 11.7 mm diameter. For economic reasons, the total thickness of the radial blanket has been chosen to correspond to only two rows of elements. The resulting breeding ratio of the Mark Ia core is only around 1.0. With a different core configuration, a breeding ratio of about 1.2 could be achieved.

Fuel reloading will take place once a year and for this purpose, the reactor must be shut down. By means of a lifting device on the innermost of the triple-rotating shield plugs, fuel elements may be removed to a storage outside the core. From there they are extracted within a sodium-filled can and transported to a sodium-cooled storage tank in the reactor building. Another gas-cooled storage tank serves to store fuel elements after a cooling-down period, and blanket elements. Both tanks combined can hold about 1 1/2 cores, including blanket elements.

During refueling the control-rod drives must be disconnected from the inserted absorbers.

The absorber material of the control and shut-down systems is boron carbide (B_4C). The first shut-down system operates with absorber rods actuated by de-energization of control-rod scram magnets, the rods being rapidly inserted by gravity. The second shut-down system makes use of articulated absorber modules that form flexible chains below the core region. They are designed to be pulled up in an emergency. This design makes for redundancy and diversity of the shut-down system.

The mechanical structure of the core provides for a restrained core.

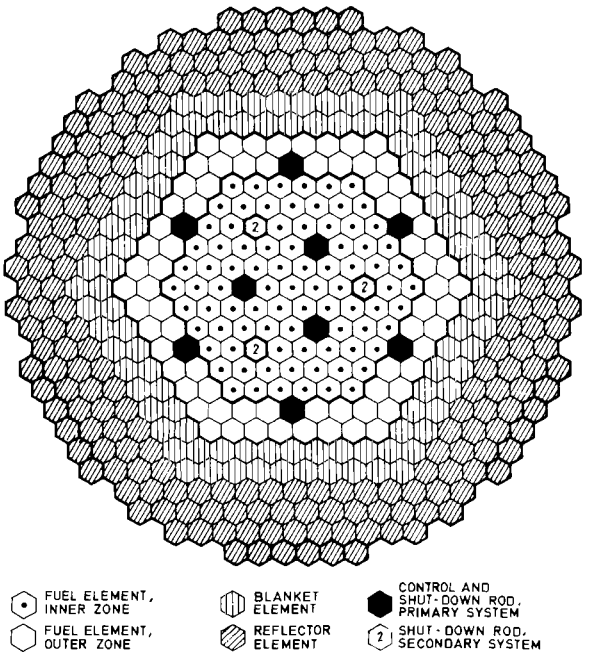


Figure IV-1: Cross Section of the SNR 300 Reactor Core

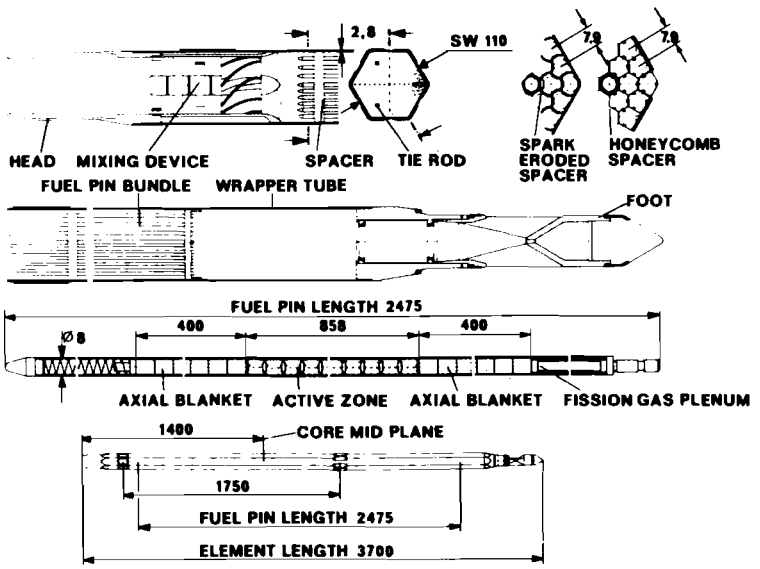
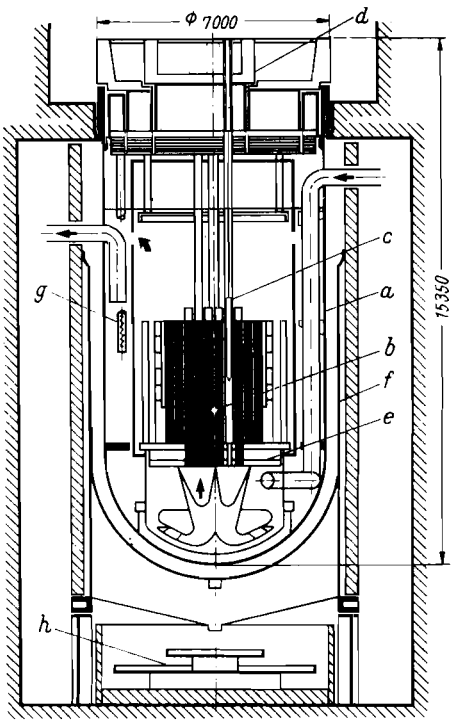


Figure IV-2: SNR 300 Mark Ia Fuel Element (unit: mm)

1.1.3 The Primary Heat Transfer System

The reactor core is enclosed by a double-wall vessel (see Figure IV-3) of 6.7 m diameter and 14.8 m height, protected by a thermal shield from heat and radiation of the core. The vessel and its shield plug are suspended from the upper part of the concrete structure of the reactor cell via bolts and a steel-support girder. The concrete and steel structures are designed to withstand up to 12,000 t of dynamic loads that may result in the unlikely event of a core disruptive accident.

Under normal operating conditions, the reactor vessel is supposed to withstand low pressure only because the sodium to be used for heat transfer is liquid, its saturation point being close to 900°C, which is very high above the mean outlet temperature of 546°C. The sodium is made to flow upward through the core at a velocity of about 7 m/sec. The total coolant flow is 3.5 t/sec.



a REACTOR VESSEL
b REACTOR CORE
c SHUT-DOWN ROD
d ROTATING PLUG SYSTEM
e CORE-SUPPORT STRUCTURE
f DOUBLE-WALL VESSEL
g DIVING COOLER OF EMERGENCY CORE-COOLING SYSTEM
h CORE-RETENTION SYSTEM

Figure IV-3: Vertical Section of SNR 300 Reactor

The sodium outlet and inlet openings of the vessel are located high enough so that the reactor core remains immersed in sodium even in the case of a rupture in the pipes of the primary system. These pipes are of a diameter of about 0.60 m. They are embedded in steel cavities shaped such that the sodium leak is as small as possible.

A primary sodium pump (see Figure IV-4) is positioned in the hot leg of each of the three primary loops. Each primary pump is designed for a capacity of 5000 m^3/h. All the sodium pumps of the SNR 300 are of the centrifugal (as opposed to the electromagnetic) type.

Having left one of the three primary pumps, the hot sodium is passed on to an intermediate heat exchanger (see Figure IV-5). It is of modular design; each loop contains three such heat exchanger modules with floating lower heads. The primary sodium passes downward through the module, heating up the secondary sodium which, in a counterflow, is flowing through a straight tube bundle. Special precautions are taken to protect the intermediate heat exchangers and other sodium components from thermo-shock effects, which are due to sudden temperature changes of the sodium (whose heat conductivity is excellent).

Within the components of the primary system, free sodium surfaces are protected from chemical reactions by means of argon as a cover gas. Argon also serves to buffer the vessel head and pump seals from sodium precipitation. A gas-bubble separator prevents gas bubbles from being carried into the core by the reentering sodium coolant. The entire primary cell is inertized by nitrogen gas. All of the primary heat-transport system is encased by the inner containment that shields the service floor and other accessible parts of the reactor building from the radioactivity of the primary system.

1.1.4 Steam Generation

Steam is generated by the secondary sodium in two steps. In the evaporators the feed water, after preheating up to 252°C, is evaporated. In a second step, the saturated steam is superheated up to approximately 500°C. The evaporators have the same principal design (once-through evaporator, see Figure IV-6) as the superheaters. For two of the tree loops, a straight-tube design is envisaged; for the third loop, a helical-tube design will probably be used so that a wider range of experience can be gained. The straight-tube design makes it easier to localize tube leaks, but raises additional welding problems. Each module is designed for 85 MW(th).

Possible sodium-water reactions are a chief concern about the use of steam generators in a sodium-cooled reactor. The precautions taken against them include a leak-detection system (based on hydrogen detection), rupture discs in connection with a pressure relief system, and cyclones for external combustion

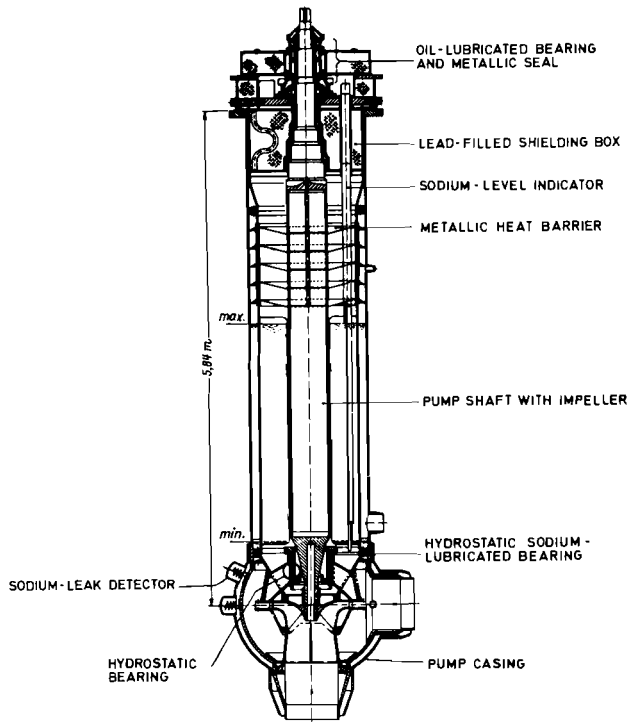


Figure IV-4:
SNR 300 Sodium Pump

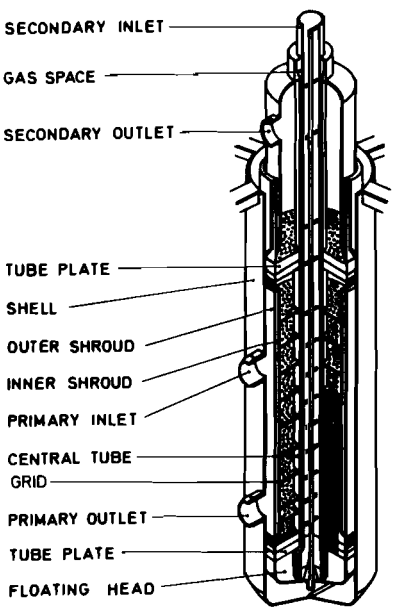


Figure IV-5:
SNR 300 Intermediate Heat Exchanger

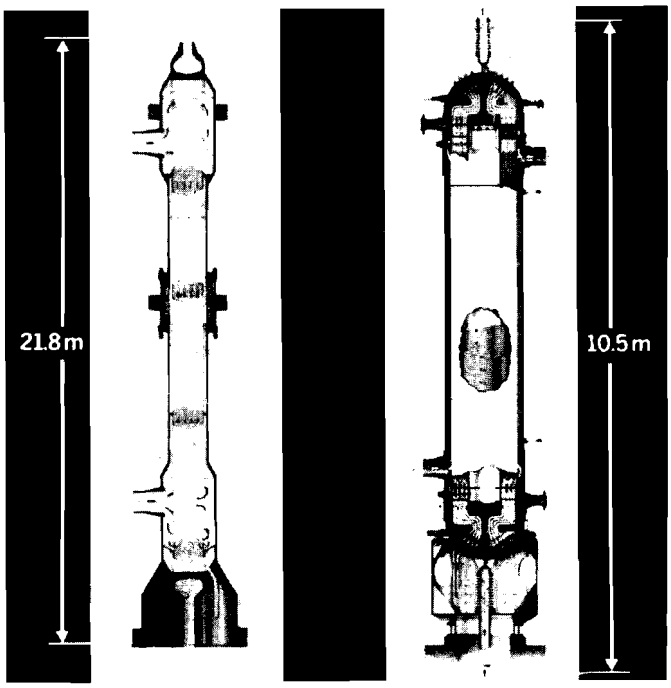


Figure IV-6: SNR 300 Steam Generators

of reaction products. Three separate steam-generator buildings provide for "spatial redundancy"

The single shaft 3000 rpm turbine (for high, medium and low pressure steam) and the generator are conventional technology. As in conventional power stations, steam is condensed after passing the turbine, and then used as feed-water in a closed water cycle. The steam-cooling system includes a cooling tower.

1.1.5 Materials Selection

The canning material for the thin tubes of the fuel rods is one of the most important materials to be selected in fast-reactor design. Canning must withstand high temperatures (700°C hot spot temperature) and high neutron flux and fluence, which lead to embrittlement, swelling, and creep of the material. Creep

phenomena are also induced by fission-gas pressure and fuel swelling from inside the rod. With increasing burn-up, chemical interactions between fuel and canning come to have an important effect on the material. For this reason, the oxygen/metal ratio of the fuel must be kept low. On the outside, the canning tubes are subject to corrosion from impurities of the sodium coolant. All these phenomena had to be taken into account in the specifications of the SNR 300 for fuel, canning, sub-assemblies and the sodium cleaning circuit design.

For the canning of SNR 300 fuel rods, "Werkstoff-Nr. 1.4970" (X10 CrNiMoTiB 15 15) has been selected, a high temperature austenitic steel stabilized by titanium. A special pretreatment (solution treating and cold working) is to render additional advantages with respect to swelling under irradiation. The Ti-stabilization is to diminish carburization of the grain boundaries by C-impurities which is due to mass transport in sodium.

For the reactor vessel and the components of the primary and intermediate heat transport systems, the German austenitic steel "Werkstoff-Nr. 1.4948" has been selected. It corresponds to AISI SS 304. This material is being tested for its strength under hypothetical accident conditions, including strains beyond the elastic range.

For the steam generators, the Nb-stabilized ferritic steel "Werkstoff=Nr. 1.6770" (2 1/4 Cr - 1 MoNb) is to be used as tube material for the evaporators and superheaters.

For the inner shielding of the reactor vessel stainless steel is used, whereas serpentine concrete is used as outer shielding material. Steel and concrete are the containment materials.

1.1.6 Engineered Safety Features

Three levels of design for safety in the SNR 300 can be differentiated:

(a) Conservative inherent-safety design for high reliability under operating conditions;

(b) Protection systems to assure that non-routine events be prevented, stalled or accommodated safely; and

(c) Additional assurance of public safety even for extremely unlikely and unforeseen circumstances.

Item (a) was briefly covered in the preceding sections. It includes design features providing, for example, fuel pin integrity, coolant capability for core-heat removal, and reactivity control.

Item (b) includes a number of active and passive engineered safeguards. The two independent and diverse shut-down systems were described briefly. Their actuation is designed to effect a de-energization of the coolant pump main drives, free coast-down, and the start-up of pony motors with 5% of the nominal speed. In this way strong thermo-shocks are to be avoided.

The reactor-power control program is based on load following; the coolant flow control program is based on constant ΔT (constant outlet temperature of the coolant).

A Diesel-powered emergency supply system assures an additional power supply of the pumps.

The decay heat of the fuel after shut-down is removed through the three main coolant loops. A separate emergency cooling system serves as back-up. It is based on six coolers immersed in the reactor vessel and on natural sodium convection in the vessel. Special design features (elevated piping-guard vessel concept to limit the effect of pipe rupture; cf. preceding sections) permit that in all cases a minimum sodium level is maintained in the vessel.

A system of in-core instrumentation including automatic signal evaluation is designed for the early detection of disturbances such as coolant blockages. Each fuel subassembly has several thermo-couples to measure its sodium outlet temperature, so that the damage can be localized. Integral detection procedures include delayed neutron monitors in the primary sodium and fission product detection in the cover-gas.

Another surveillance system serves to detect leaks along the primary and intermediate coolant circuits. Sections of these loops can be shut off by valves. In the case of tube leakage individual tubes of the steam generators can be blocked so that the plant need not be completely shut down because of a single leak. Provisions for controlling sodium-water reactions have already been mentioned.

For item (c) the barrier concept is particularly important. A release of radioactivity is prevented by a series of containments. Under normal operating conditions, containment functions are fulfilled by the fuel which retains a large part of the fission products, canning, the reactor vessel, the primary coolant circuit piping and its components, and the primary sodium.

The inner and outer containments of the reactor building provide additional safety (see Figure IV-7).

The inner containment that encases the primary cells is filled with nitrogen. Its walls are completely clad with steel. It withstands a differential pressure of 0.3 bar at a leakage rate of about four per cent per day (per volume). In a pump-failure accident, heat is by natural convection trans-

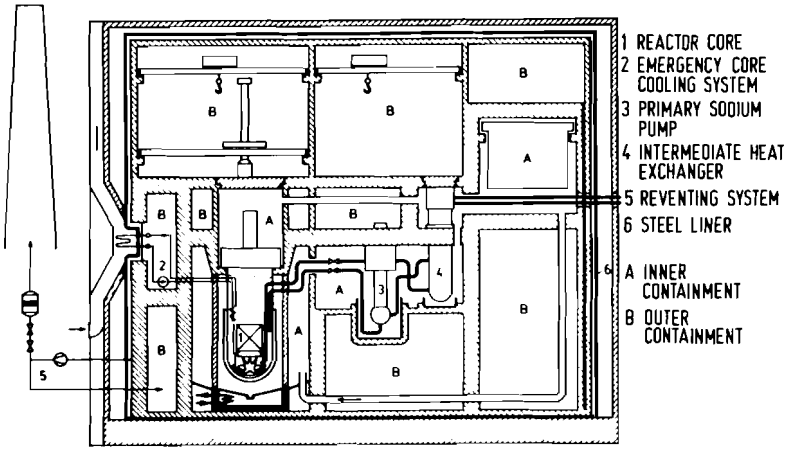


Figure IV-7: SNR 300 Containment System

ported out of the reactor and primary cells to a pressure-relief room, in which concrete blocks serve as a heat sink.

The outer containment is filled with air (the service floor being accessible during operation), and it is designed to withstand a differential pressure of 0.25 bar. It is completely surrounded by a steel liner. The gap between the concrete walls and the liner can be revented into the containment for some days after an accident.

The reactor building is of rectangular shape, 98 m in length, 58 m in width, and 57 m in height. It is designed to withstand external hazards such as airplane crashes (a phantom fighter plane), earthquakes, and explosion of gas clouds (after a boat collision on the Rhine river). Therefore, the heavily armed outer walls are of a thickness of up to 1.2 m. The emergency power-supply building is similarly protected.

The SNR 300 will also be equipped with a core-catcher (see Figure IV-8) to control the consequences of a serious melt-down accident. This core-catcher is to collect and cool the fuel debris that might penetrate the reactor vessel after fuel melting in an accident involving the entire core, when all internal coolant systems fail. The core-catcher is made up of a steel cavity of almost 12 m in diameter that serves to collect the sodium contained in the vessel; the debris is caught by a layer of uranium oxide or thorium oxide bricks; the NaK cooling system underneath this layer serves to thermally insulate the concrete of the foundation plate.

The containments and the core-catcher should be viewed especially in connection with a--hypothetical--Bethe-Tait

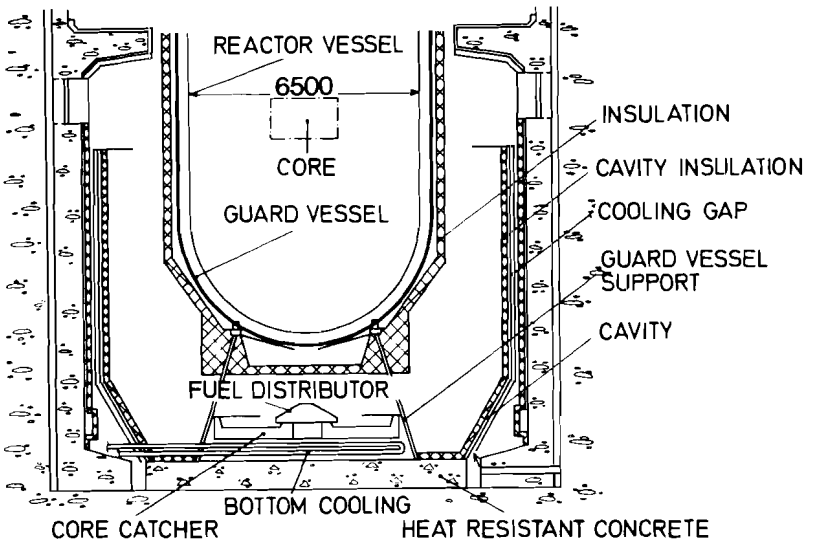


Figure IV-8: SNR 300 Core Catcher

accident, where mechanical energy of up to 370 MWsec is released. The vessel has been laid out to withstand this mechanical pressure. If--after all heat removal systems and emergency heat removal systems have failed--the fuel melts down through the vessel into the core-catcher, pressure and temperature within the inner containment will gradually be rising. Since the containment is not completely leak-tight, a certain amount of aerosols and gases and of enthalpy may be transferred to the space in the outer containment. This extreme case would require reventing, and, possibly, a controlled release of the containment atmosphere to the stack a few days later.

1.2 Fast Breeder Prototype Reactors BN 350 and BN 600

(A.M. Belostotsky)

1.2.1 General Characteristics of the Plants

The BN 350 nuclear power plant has been built in Shevchenko on the shore of the Caspian Sea. The thermal power output of this plant is 1000 MW(th). The bulk of this output is used to desalinate water from the Caspian Sea, with the remainder to generate electric energy.

During nominal operation the plant will support three turbogenerators of 50 MW each, and a desalination complex with a total output of 120,000 t/day of water.

Construction of the Shevchenko nuclear plant was completed toward the end of 1972. In November 1972 the reactor became critical; on May 16, 1973 it began to produce electricity, and steam was released into the desalination complex. This experience in building the world's largest experimental-industrial nuclear power plant with a fast-breeder reactor, the BN 350, permitted to proceed to the third block of the "Kurchatov Bjelojarsk" nuclear power plant: in 1968 in the Urals construction of the BN 600 was started, an industrial nuclear power plant with a capacity of 600 MW(e), [IV-1].

1.2.2 Basic Technical Considerations for BN 350 and BN 600

At both plants, reactor heat is removed by means of a three-circuit scheme: liquid sodium in the first and second loops, water and steam in the third. The basic specifications for these plants are given in Table IV-I.

In order to transfer heat to the steam generators, the BN 350 is equipped with six parallel loops for the first circuit and six independent loops for the second circuit; of these, one loop of each circuit is a reserve loop.

Having passed the turbines the steam enters the desalination complex at a pressure of 6 kp/cm^2. Then at a temperature of approximately 100^{0}C it flows as a condensate into a high-pressure pre-heater and deaerator, and then by means of feed-water pumps back to the steam generators. This BN 350 heat-transfer system is schematically depicted in Figure IV-9.

The equipment of the BN 350 reactor installation is spatially separated: it has five operating loops in the first circuit; each loop of the first circuit is located in its own concrete box, which is again isolated from the reactor pit and neighboring containments of other loops, by heavy, protective concrete walls. It is the main condition for the arrangement of the first circuit to lower the neutron activation of the coolant by segregation in the intermediate heat exchangers so that the sodium in the second circuit is not activated.

This arrangement involving segregation of first circuit subassemblies simplifies replacement or repair of equipment which has been exposed to high levels of radiation; it allows one to shut off any loop by means of valves located on inlet and outlet pipes, and to carry out repairs after it has been dried of radioactive water.

Sodium circulation of the first and second circuits is realized by means of centrifugal pumps of the console type. The pressure of argon in the gas cavities of the reactor and the pressure in the pumps of the first circuit is 1.9 kp/cm^2. Each loop of the first circuit is equipped with two gate valves mounted on the outlet and inlet pipes.

Table IV-I: Basic Specifications of BN 350 and BN 600 Fast Breeder Power Plants

		BN 350	BN 600
Thermal Power	MW(th)	1000	1480
Electric Power	MW(e)	350*	600
Burn-up	MW(th)·day/t	50,000	90,000
Coolant		Na	Na
Coolant Flow Rate for One Loop of the First Circuit	m^3/h	3200	9300
Coolant Temperature			
Core inlet	^0C	300	380
Core outlet	^0C	500	550
Steam Temperature at Turbine Inlet	^0C	440	510
Steam Pressure at Turbine Inlet	kg/cm^2	50	140
Operational Time Between Reloading of Fuel	days	50	150
Core Dimensions			
Height	cm	106	75
Outside Diameter	cm	150	205
Number of Control and Safety Rods		12	27

* or 150 MW(e) and 120,000 t/day desalinated water.

Fuel assemblies of the core and blankets are installed in a pressurized collector mounted in the pressurized chamber of the reactor tank. The core consists of 200 fuel assemblies filled with fuel elements that contain uranium oxide and breeding material. Within the core are located two automatic control rods, six rods for burn-up compensation, three safety rods, and one rod for temperature-effect compensation. The core along its axial surfaces and perimeter is surrounded by a blanket zone of depleted uranium. The blanket at the ends is part of the core assemblies, the radial part is formed by the fuel assemblies containing the fuel elements of the blanket.

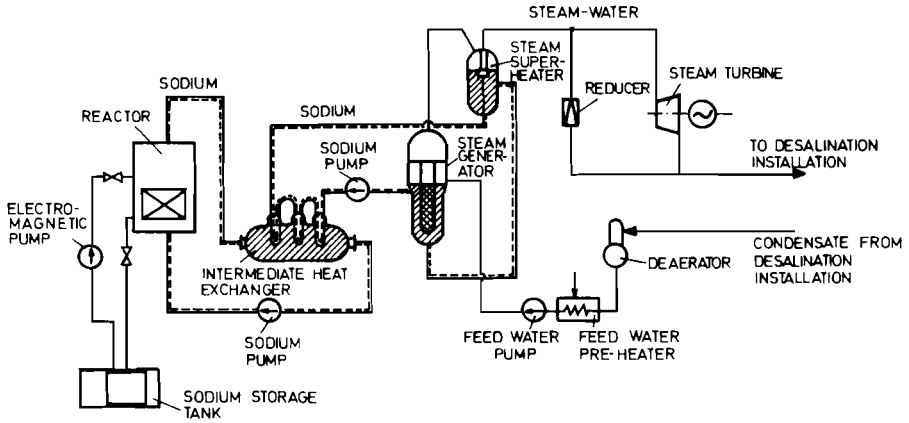


Figure IV-9: BN 350 Heat Transfer System

The reactor tank, the most crucial part of the entire BN 350 assembly, is a vessel of varying diameter (its largest diameter is 6000 mm) of stainless austenitic steel. It is cooled by sodium passing through the gap between the walls of the reactor tank itself and the heat shield. Cooling provides for an operating temperature that does not exceed 450°C.

The tank temperature in the area of the output pipes, at nominal operating capacity, is 375°C. The maximum temperature of the reactor tank at its highest part is 420°C.

The lower part of the tank forms a pressure chamber, into which sodium enters through pipes from the pumps. While flowing from the bottom up through the core and the blanket region into a repository, the sodium is heated; then it flows through the upper mixing chamber of the tank, and via pipes passes into the heat exchangers. To avoid leakage of the sodium in the case of depressurization of the main tank, the latter is enclosed in a containment.

The internal surfaces of the tank and output pipes have shields which lower temperature stresses in the case of a sudden coolant temperature change. Cooling of the tank, as was already said, is provided for by "cold" sodium. As material for biological shielding outside the reactor, iron-ore concentrate, graphite, steel, and concrete are used.

The upper part of the tank serves as support for two rotating plugs which provide the reloading mechanism for fuel and blanket assemblies. The rotating plugs also serve as an upper biological shield; for sealing them, hydraulic seals with an eutectic alloy are employed.

Loading and unloading of fuel assemblies is carried out by means of a loading mechanism which is installed on the small rotating plug, and which transfers fuel assemblies within the reactor.

Transport of fuel elements from the core takes place via a closed duct from the reactor to an external storage area. Reloading is carried out after a maximum fuel burn-up of 50,000 MW(th)·day/t has been achieved. By means of the transfer mechanism a fuel element is removed from the core to a compartment of the unloading elevator. The transport elevator is a sloping lift with a cage containing a compartment for fuel elements. In this cage, the fuel element is moved upwards to the next fuel-element-transfer mechanism. The mechanism clasps the element, lifts and moves it to a transfer box. Then it is transported to a cylinder for spent fuel elements.

Spent fuel elements from the cylinder enter washing compartments, and after washing, pass into a basin for fuel element unloading. The transportation of the fuel elements in boxes takes place in an argon environment. Control of the fuel element transfer mechanism is completely automatic. The cycle for transferring a fuel element from the core to the cylinder is forty minutes.

Figure IV-10 shows a vertical section of the BN 350 reactor configuration.

Compared to the BN 350, the BN 600 has a higher thermal and, correspondingly, electric power. Burn-up has also been increased (100,000 MW(th)·day/t instead of 50,000 for the BN 350). The operational time between fuel-element replacements has been greatly increased (150 days instead of 50), and the temperature of the sodium leaving the reactor has been raised (550^{0}C instead of 500^{0}C). All of this has led to optimal steam parameters; steam temperature and pressure at turbine intake are 540^{0}C (instead of 440^{0}C) and 140 kp/cm^2 (instead of 50 kp/cm^2), respectively. Such high steam parameters have permitted the use of turbines of conventional design. At the BN 600 plant three turbo-generators will be installed, each with an electrical power of 200 MW.

For the BN 600 a more economical and, from an engineering point of view, successful solution has been achieved by arranging the first circuit in a pool. Core, pumps and intermediate heat exchangers are located in a single container. This integral arrangement was first realized in the USSR with the construction of large nuclear power plants with fast breeder reactors. Figure IV-11 shows a comparison of the BN 350 loop-type system and the BN 600 pool-type system.

Cooling is carried out in a three-circuit scheme: in the first and second circuits by means of liquid sodium, in the third by means of water and steam. Sodium of the first circuit is cooled in intermediate heat-exchangers by the sodium

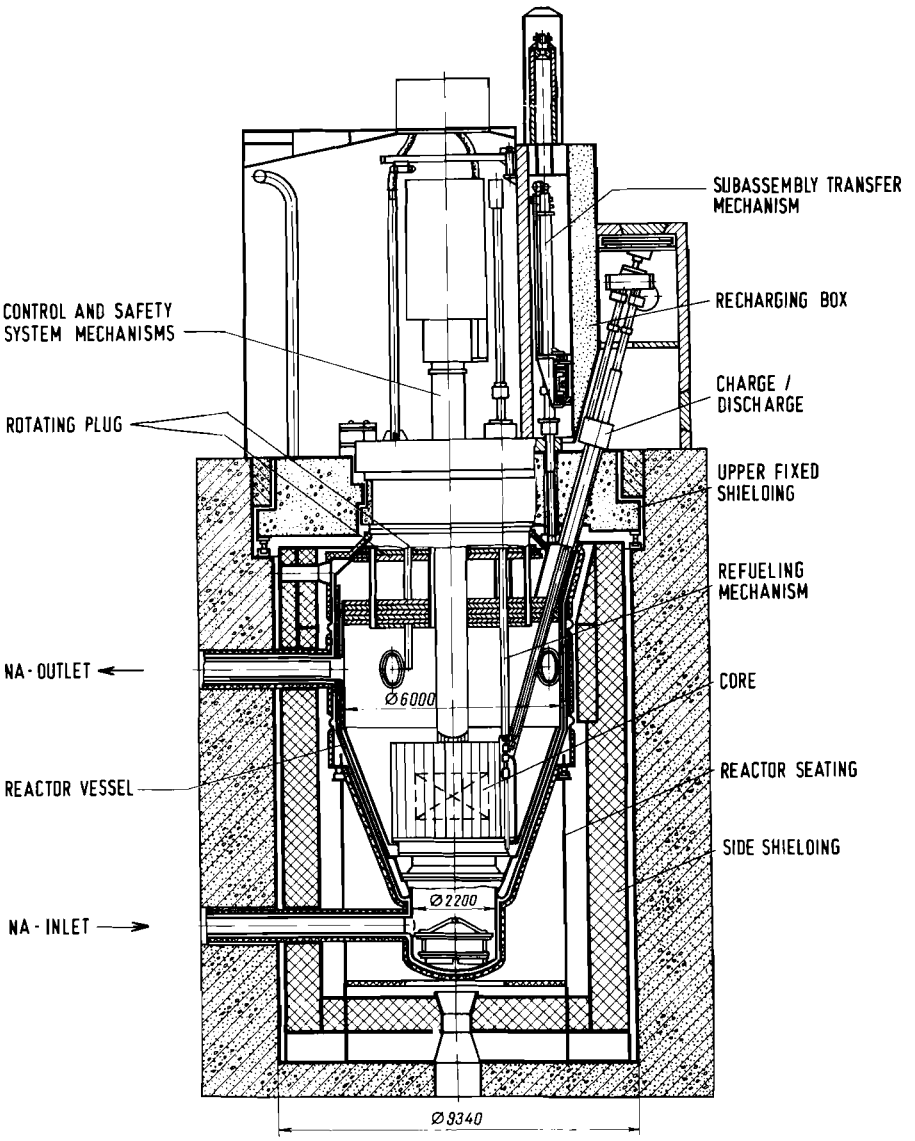


Figure IV-10: BN 350 Reactor Configuration

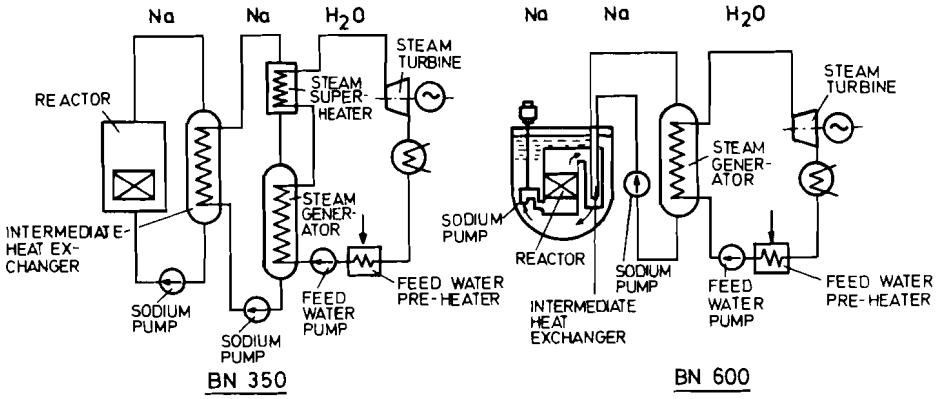


Figure IV-11: BN 350 Loop Type System vs. BN 600 Pool Type System

of the second circuit. Heat from the intermediate heat exchangers is conducted to steam generators via three independent loops of the second circuit.

Core, pumps, intermediate heat-exchangers, and biological shielding are located in the reactor tank. The coolant of the first circuit moves through three parallel loops, each of which includes two heat-exchangers and a centrifugal circulation pump.

The core and blankets are mounted on a support grid where coolant is distributed within the fuel assemblies along their respective fuel elements. The core consists of 370 fuel assemblies containing nuclear fuel--uranium oxide and breeding material--and 27 control and safety rods. The control and safety system consists of two automatic control rods, 19 burn-up and temperature-effect compensators, and six emergency safety rods. Along its perimeter and axial surfaces, the core is surrounded by a blanket zone consisting of fuel elements filled with depleted uranium dioxide.

The reactor tank is a cylindrical container; it has an eliptical base and a conical upper part in which the rotating plugs are installed. The main equipment of the first circuit is mounted in the tank: the reactor support grid with the fuel elements of the core, blanket zone, radial neutron shield, intermediate heat-exchangers, and circulation pumps of the first circuit. The reactor's weight is transferred via its support ring to supports anchored on a foundation plate.

Biological shielding consists of cylindrical steel plates, steel billets and pipes with graphite filler. The reactor container is encased in a protective shield.

The upper part of the tank serves as support for the rotating plug and rotating column which provide for fuel element loading. The rotating plug and column also function as a biological shield.

Loading and unloading of fuel assemblies is carried out by a complex of mechanisms; it consists of two loading mechanisms mounted in the rotating column, two elevators for loading and unloading, and a revolving-type transfer mechanism located in a hermetic box. Figure IV-12 shows the BN 600 reactor configuration.

Despite major changes, there were many similar aspects in the construction of the BN 350 and BN 600 reactors. The seals for the rotating plugs, loading and hoisting mechanisms, the basic assemblies for fuel element transfer, fuel element design, etc., remained essentially unchanged. Thus experience gained in building the BN 350 was fully utilized in the construction of the BN 600.

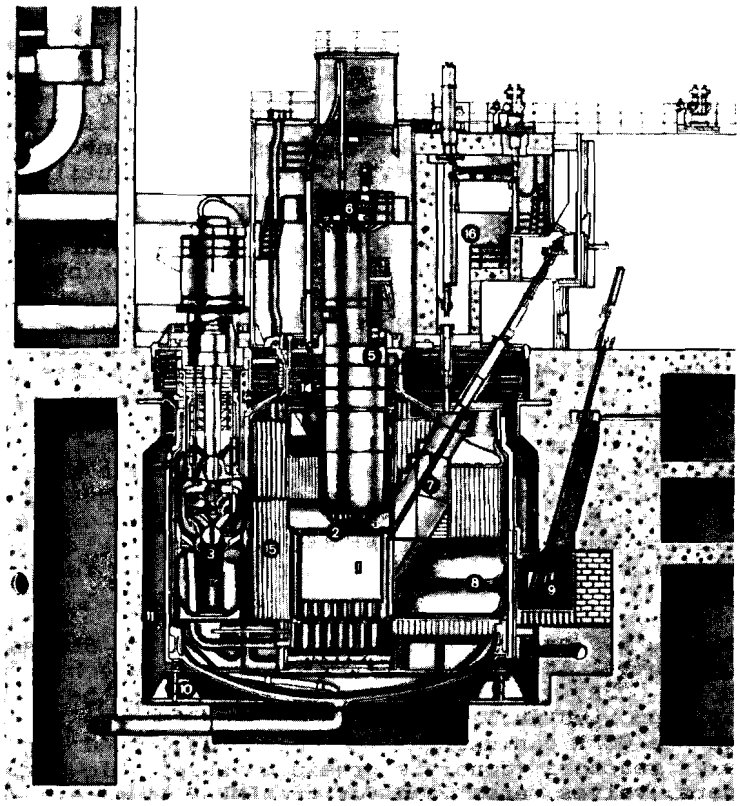
One of the important conditions for successful and safe reactor operation is the reliability of its core from a thermal engineering standpoint. Monitoring of basic thermal parameters and physical conditions in the reactor core plays a crucial role. Any deviation from normal operation must be detected. In the BN 600, besides monitoring of neutron production rate, its change and reactivity, there is also surveillance for:

- delayed neutrons penetrating into the coolant;
- the presence of fission products in the reactor cover-gas (resulting from their escape from damaged fuel elements);
- flow rate variations of coolant flow through the reactor are measured by electro-magnetic flowmeters; and
- variations in coolant temperature upon exit from the reactor.

If any deviation from normal operating conditions is detected the reactor power is reduced, or the reactor is completely shut down.

If, after reactor shut-down, a defective fuel element is detected its removal is anticipated by means of the so-called defective fuel element detection system. This system consists of a mechanism installed in the rotating plugs which can reach any element in the core which might contain a leaky fuel pin in order to remove the element from the reactor.

With the accumulation of operational experience with the large experimental-industrial reactors BN 350 and BN 600 through nuclear plants of this type, a wide road will be opened toward the development of a large-scale Soviet nuclear power industry.



1 REACTOR CORE
2 FUEL ELEMENTS
3 SODIUM PUMP
5 ROTATING-PLUG SYSTEM
6 DRIVE MECHANISM FOR CONTROL AND SHUT-DOWN SYSTEM
7 TRANSFER SYSTEM FOR IRRADIATED FUEL ELEMENTS
8 NEUTRON DETECTORS
9 BLOCK OF IONIZATION CHAMBERS
10 REACTOR-CORE SUPPORT STRUCTURE
11 REACTOR-CELL
14 ROTATING PLUG
15 SHIELDING STRUCTURE PROTECTING VESSEL FROM NEUTRON RADIATION
16 FUEL-ELEMENT TRANSFER CELL

Figure IV-12: BN 600 Reactor Configuration

1.3 Other Fission Breeder Reactor Systems

1.3.1 The Gas-Cooled Fast Breeder Reactor (GCFR)

Similar to the case of thermal reactors, with several different types being on the threshold of commercial availability, there are technological and industrial reasons to develop an alternative fast breeder system. Particularly, if one considers that fast breeders will be the most important energy system beyond the year 2000. It is hard to believe that utilities would then rely on one fast breeder system only.

When the development of fast reactors was started, the only feasible coolant appeared to be a liquid metal which is capable of cooling small cores of very high power densities. The increase in size of the envisaged nuclear power plants and the change from metallic to oxide fuels have made the requirements on the heat transfer capabilities of the coolant less stringent. Furthermore, in the meantime, the technological development of gas-cooled thermal reactors (MAGNOX, AGR, HTGR) has considerably improved the obtainable thermal performance of gas coolant. The technological progress that could be of advantage here is the development and successful operation of very large gas blowers, of prestressed-concrete pressure vessels capable of containing high pressure gas, and of the so-called "artifical roughness" for a better heat transfer at the surface of the fuel element pins.

To date two private companies in the United States and in the FRG (General Atomic and Kraftwerk Union) have had teams work on the GCFR, with the active assistance of the governmentally supported nuclear centers in each country [IV-2, IV-3]. Although the GCFR can use the technology developed for thermal gas-cooled reactors (concrete pressure vessel, blowers, heat exchangers, valves, etc.) and for the LMFBR (fuel elements, neutron physics), the costs of developing this new reactor type would still be very high. Therefore, the development programs of this alternative fast breeder system are progressing more slowly than those of the presently favored LMFBR system.

Table IV-II shows major performance data of the 1000 MW(e) GCFR design of Kraftwerk Union. The very good breeding ratio (1.45) and the relatively high fissile plutonium inventory (3.4 kg/MW(e)) can yield a fissile plutonium doubling time in the range of 11 to 12 years.

Contrary to liquid-metal-cooled fast breeder reactors, gas-cooled fast reactors are designed to operate under a helium gas pressure of 80 to 120 atm. Helium gas is pumped downwards by blowers through the core fuel elements and increases its temperature from 270 to 550 ^{0}C. The fuel rods will have an artifical roughness on the surface in order to improve the heat transfer. The core, the gas blowers, and the heat exchangers are located within a prestressed-concrete vessel (see Figure IV-13).

Table IV-II: Major Performance Data of a 1000 MW(e) Gas-Cooled Fast Reactor (Design Kraftwerk Union, FRG)

		1000 MW(e) GCFR
Reactor thermal power	MW(th)	2780
Net electrical power	MW(e)	1030
Plant efficiency	%	37
Maximum fuel burn-up	MW(th)·day/t	100,000
Breeding ratio		1.45
Average enrichment	%	12.1
Fissile plutonium inventory	kg	3450
Power rating, core	MW(th)/kg	0.76
Fuel lifetime	yr	3
Doppler constant	$T \cdot \frac{dk}{dT}$	$-6 \cdot 10^{-3}$
Reactivity swing between refueling	$	-0.4
Helium reactivity worth	$	1.2
Core volume	m^3	10.2
Reactor-coolant pressure	atm	120
Clad hot-spot temperature	^0C	700
Pin diameter	mm	8.2
Helium, reactor inlet temperature	^0C	273
Helium, reactor outlet temperature	^0C	555
Main turbine steam conditions	^0C/atm	510/100

Safety problems of a GCFR are of a different nature than those pertinent to an LMFBR; they are essentially connected with a fast loss of coolant pressure or a failure of the cooling blowers [IV-4].

1.3.2 Thermal Breeder Reactors

The principle of breeding is not restricted to the use of fast neutrons. The fundamental condition for the feasibility of breeding is $\eta > 2$, if η is the number of neutrons released per

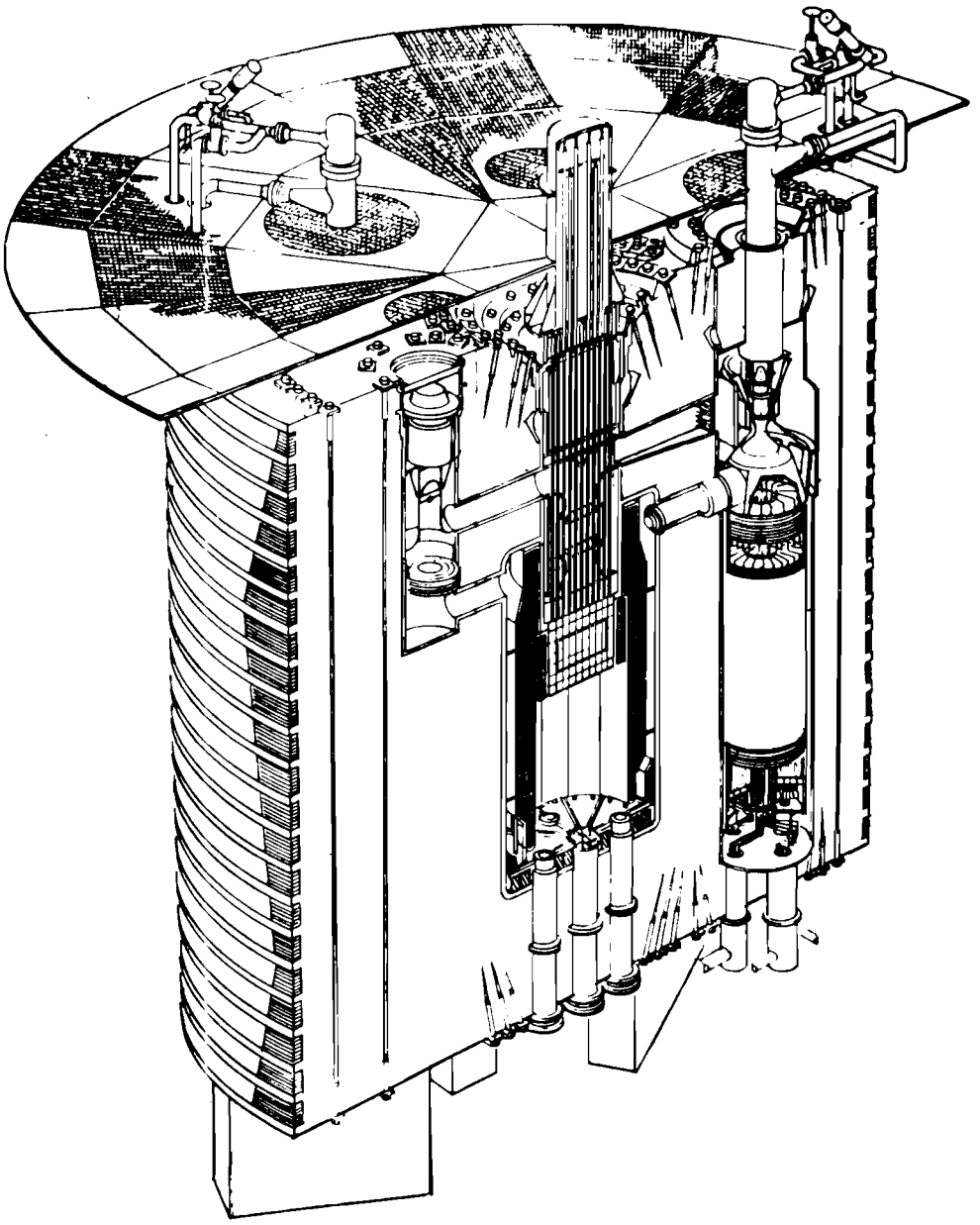


Figure IV-13: Nuclear Steam Supply System of the 300 MW(e) Gas-Cooled Fast Reactor Demonstration Plant (General Atomic, San Diego)

neutron absorbed [(n,γ) and (n,f)] in the fissionable atom. $\eta > 2$, on the one hand, points to the use of Pu239 and Pu241 in a fast spectrum. It must be a hard spectrum because η falls off quickly with decreasing energy of the fissioning neutrons. And in terms of physics, the problem of the fast breeder is to keep the energy of the fissioning neutrons sufficiently high. On the other hand, $\eta > 2$ points to U233 in a thermal spectrum and, accordingly, it is the U233-Th fuel cycle which is the prime candidate for a thermal breeder. As far as U233 is concerned, it therefore is the problem of thermal breeders to have the energy of the fissioning neutrons sufficiently thermalized. It must be noted, however, that all forms of mixed fuel cycles can be and have been considered. For instance, U233 is feasible for the use in fast breeders, and Pu has been considered for the use in thermal breeders. The simple reactor physics reasoning as given above then becomes much more complex.

Basically there were two reasons that led to the preference of fast breeders:

(a) The ease of meeting the condition $\eta > 2$, even in the presence of engineering constraints;

(b) The compatibility of the (U,Pu) fast breeder fuel cycle with those of existing thermal nuclear power reactor stations, principally the LWR.

One should, nevertheless, note that there continues to be a potential for thermal breeder reactors*. The two key issues deriving from the condition $\eta > 2$ are fission product removal and Pa233 poisoning. Enhanced fission-product removal necessitates either frequent refueling and, consequently, high fuel-cycle costs, or the introduction of the homogeneous-reactor concept with (inherent) continuous fission-product removal. Pa233 poisoning is a point because of the relative long half-life of 27.4 days of Pa233; the half-life of Np239 on the other hand, the logical analogue for the U-Pu cycle, is much shorter (2.33 days). Such poisoning in principle requires the spatial partition of U233 as fuel from the fertile material Th232, that is, the distinction between core and blanket. The other principal means to meet the problem of Pa233 poisoning is a low neutron flux, and that is a low specific power. It obviously hurt the fuel cycle economy, too.

In a large number of reactor-design studies an attempt has been made to circumvent the problems and foster the positive features of thermal breeders.

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* The chief pioneers for the thermal-breeder concept have been E. Wigner and A. Weinberg in the US; in particular, experiments and design work were carried out by the Oak Ridge National Laboratory (ORNL).

There have been four approaches to a homogenous-reactor design:

(1) Heavy-water/uranium-aqueous-solution reactor;

(2) Liquid-bismuth reactor;

(3) Heavy-water-slurry reactor;

(4) Molten-salt reactor.

(1) and *(2)* led to significant technological problems and problems of chemical stability at higher temperatures; they were no longer pursued.

The slurry concept *(3)* avoids these difficulties by using small and chemically inert ceramic particles. It was mostly pursued by F.F. Went at the KEMA laboratories, Netherlands. A 1 MW(th) experimental reactor has been in successful operation for a number of years, demonstrating the principal feasibility of this concept and, particularly, its remarkable operational stability.

The molten-salt-reactor concept *(4)* has mostly been pursued by the Oak Ridge National Laboratory (ORNL). Very significant was the successful operation of their 8 MW(th) Molten Salt Reactor Experiment (MSRE). Using a variety of molten salts it permits significantly higher temperatures than *(1)* through *(3)*. It requires the integration of continuous on-line reprocessing, which makes it very similar to the fission concept with on-line tritium reprocessing.

The three most important lines for a heterogenous thermal breeder are the following:

(5) U233-Th/heavy water reactor;

(6) High-temperature gas-cooled reactor (HTGR);

(7) Light-water breeder reactor (LWBR).

In all cases, the fundamental idea is to use existing lines of thermal-power reactor stations with already partly or fully developed engineering. It is a consistent and painful experience that it is not so much the difficulties of a principal kind which a new reactor development line runs into but rather numerous engineering difficulties; and it is not that they are unsurmountable, but they are expensive to overcome in terms of both money and time. In view of this experience it is indeed a basic advantage if already existing engineering developments can be employed. The price to be paid is in terms of reactor physics: The breeding ratio is always hurt, and is often reduced to one or even below one. Consequently, the reactor lines *(5)* through *(7)* are often referred to as near-breeders.

To obtain a near-breeder with the heavy-water/natural uranium reactor concept it is necessary to introduce the U233-Th fuel cycle and frequent refueling. Both are an economic burden.

To explore the high-temperature line, the coated-particle gas-cooled reactor was investigated in greater depth. The (n,2n) effect of Be9 was often considered with respect to enhanced neutron production. As a subsequent reaction of neutron activation (n,α), however, Li6 is produced, which is a strong neutron absorber. At the same time embrittlement of Be takes place on account of helium formation. Therefore, the idea of engaging such neutron enhancers has essentially been dropped. The attainable breeding ratio (conversion ratio) is a function of the acceptable fuel-cycle penalties. Under somewhat realistic conditions (which are hard to specify in the absence of a relating and technically existing fuel cycle) one may expect a "breeding ratio" of 0.8 only.

The other possiblility is to engage the engineering of the Pressurized Water Reactor (PWR), which has been extraordinarily successful. This has mainly been pursued in the US by Bettis Atomic Power Laboratory. The idea is only to change the core arrangement, following the seed and blanket approach. It provides for fuel elements of mainly fissionable material (seed) in an annulus of fertile material (blanket). The seed part of the fuel element is expected to be movable upward and downward, thus allowing for reactivity control without parasitic absorption in control rods. The clustering of fuel in the seed and blanket arrangement is meant to enhance epithermal and fast fissions by providing an extremely small coolant-water fraction in the fuel element. Designers expect a breeding ratio of one under operating conditions.

As mentioned before, the nuclear fission reactor community has concentrated on the fast breeder, because there the choice between sound engineering and good reactor physics is less painful. But one must realize that the potential for thermal breeding continues to exist. If thermal fission breeders turned out to be a necessity or of sufficient commercial interest, the most promising candidate would probably be the molten-salt reactor. The reactor type is not pursued any more because of lack of funds, the absence of an industry that identifies itself with it, and the lack of a benefit that could otherwise not be achieved. For more details and references we draw the reader's attention to a survey article by A.M. Perry and A.M. Weinberg [IV-5].

2. FUSION REACTORS

2.1 Fusion Reactor Designs

Since the beginning of large-scale reactor designs in the early 1970s there have been at least 23 individual studies. The largest number of studies, 13, have been conducted on TOKAMAKs, while six have been on laser and electron-beam reactors, three on Mirrors, and one on Theta Pinches. A list of their designs and origins is given in Table IV-III; Table IV-IV summarizes their most salient features. Artists' conceptions of the TOKAMAK reactors such as UWMAK-I [IV-7], UWMAK-II [IV-8], and UWMAK-III [IV-9], PPPL [IV-10], Culham [IV-11], and JAERI [IV-13] are shown in Figures IV-14 to IV-20, respectively. Figure IV-21 shows a twelve-module laser system from LASL [IV-19]; Figure IV-22 shows a possible reactor design for the Mirror [IV-24]; and Figure IV-23 is the overall plant design for the reference Theta Pinch reactor [IV-27].

Table IV-III: Summary of Laboratories Which Have Presented Complete or Semi-Complete Demonstration Power Reactor Designs Based on a Pure D-T Fuel Cycle*

Reactor Type	Laboratory	Reference
TOKAMAK	ORNL	[IV- 6]
	Univ. of Wisconsin	[IV- 7 to IV-9]
	PPPL	[IV-10]
	Culham	[IV-11, IV-12]
	JAERI	[IV-13, IV-14]
	BNL	[IV-15, IV-16]
	Ispra (EURATOM)	[IV-17]
	Khurchatov	[IV-18]
Laser	LASL	[IV-19, IV-20]
	LLL	[IV-21]
	Jülich	[IV-22]
	ORNL	[IV-23]
Mirror	LLL	[IV-24 to IV-26]
Theta Pinch	LASL-ANL	[IV-27]
E-Beam	Sandia	[IV-28]

* Excludes hybrids, fission-product burners, or non-electrical power producing designs.

Table IV-IV: Summary of Material Environments for Various D-T Fusion Reactor Designs

Type	Reactor	Power (MW(th))	First Wall Material	Average Neutron Wall Loading (MW/m^2)	Maximum First Wall Temperature (^0C)	Coolant	Cycles Per Year	Reference
TOKAMAK	ORNL	1,000	Nb-1Zr	0.5	1050	Li	$1.3 \cdot 10^5$	[IV- 6]
	UWMAK-I	5,000	SS 316	1.25	500	Li	$4.7 \cdot 10^3$	[IV- 7]
	UWMAK-II	5,000	SS 316	1.16	550	He	$4.7 \cdot 10^3$	[IV- 8]
	UWMAK-III	5,000	TZM	2.5	1000	Li	$1.2 \cdot 10^4$	[IV- 9]
	PPPL	5,546	PE16	1.8	663	He	$4.1 \cdot 10^3$	[IV-10]
	Culham-I	5,000	Nb	2.9	470	He	not stated	[IV-11]
	JAERI-I	2,000	Incoloy 800	2.0	630	He	$4.2 \cdot 10^3$	[IV-13]
	JAERI-II	2,000	Mo alloy	1.6	400-700	He	$4.2 \cdot 10^3$	[IV-14]
	BNL-I	3,125	SAP	1.0	400	He	not stated	[IV-15]
Laser	LASL-I	3,745	Nb	1.8	not stated	Li	$2.1 \cdot 10^7$	[IV-19]
	LLL	760	Nb	1.0	500-1100	Li	$2.5 \cdot 10^8$	[IV-21]
	Jülich-Saturn	5,000	Nb	2.5	850 max	He	$2.5 \cdot 10^9$	[IV-22]
	ORNL-Blascon	150	Cr Mo-steel	1.6	482	Li	$2.5 \cdot 10^6$	[IV-23]
Mirror	LLL	640	SS	1.6	≈650	Li	<10	[IV-24]
	LLL	2.030	V alloy	3.1	≈800		<10	[IV-26]
Theta Pinch	LASL-ANL	12,000	Nb	2.0	550-850	Li	$8.5 \cdot 10^6$	[IV-27]

We have chosen the TOKAMAK reactors as a reference for this study because they have been studied in greatest detail; they also seem to be the lead reactor concept in most national programs. The great number of TOKAMAK reactor designs helps to insure that the appropriate diversity will be achieved with respect to coolants, tritium inventory, radioactive materials inventory, and materials resource demands. We will attempt to develop an envelope of characteristics for all the reactors now under consideration as we assess the requirements of fusion power.

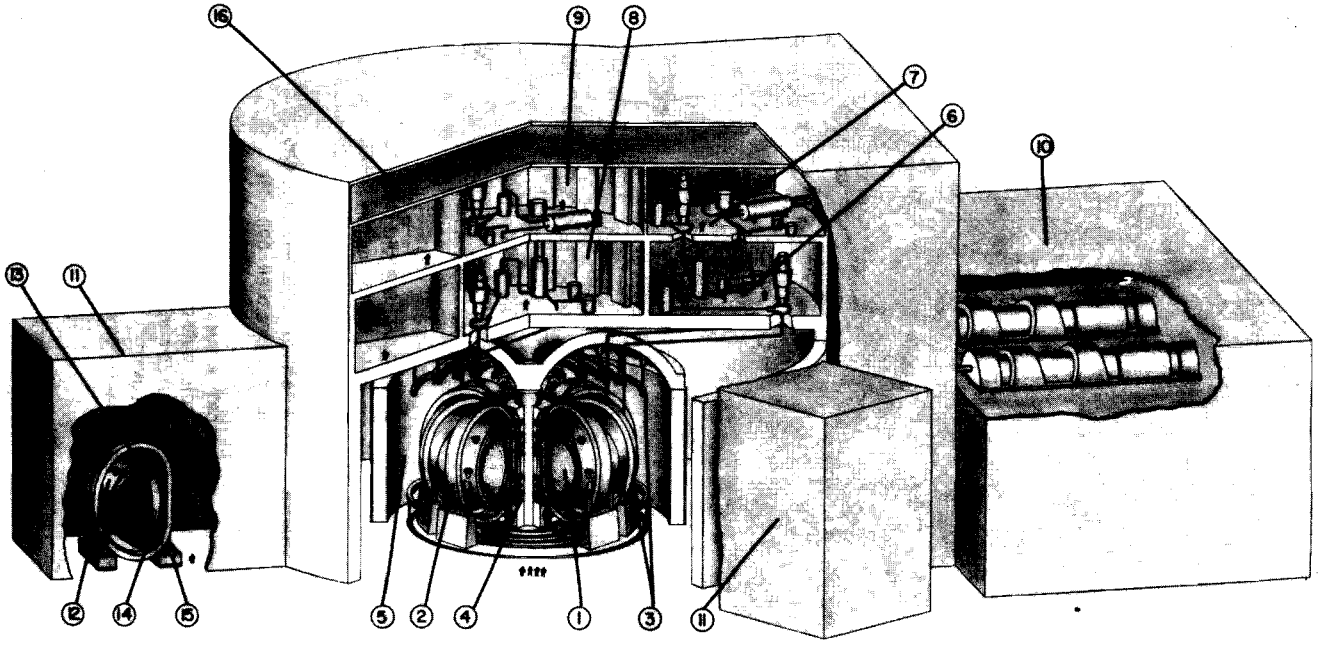
It is worthwhile to point out how the reactor designs have changed since the first full-scale reactor designs by A.P. Fraas [IV-6], and the Culham group [IV-11]. The first thing of note is the trend toward austenitic steels and, consequently, lower operating temperatures. This tends to give a different picture with respect to long-lived radioisotopes than previous articles, and it also lowers the efficiency of the reactors to that currently typical of present fission or future breeder reactors.

Another change since those early reactors is the inclusion of helium as a coolant and the potential for lower tritium inventories in the blanket by the use of solid lithium breeding coumpunds [IV-8, IV-14 to IV-16]. While this does not eliminate the problems of tritium accidents, it does tend to reduce them in the area of the blanket.

A third change is the attention paid to the balance of plant design and the inclusion of that part of the system in the materials resource demands, cost estimates, and safety hazards. This area still needs more attention, but recent studies of balance of plant designs have revealed a few surprises which were not considered in the earlier work. These are, for example, the need for load leveling systems, attention to external piping cost of high temperature systems, and the effect of magnetic field leakage [IV-7 to IV-9].

The final point is that power levels have been increased in order to find an optimum size from the standpoint of costs. Several TOKAMAK studies have fixed on the 5000 MW(th) size, while the reference Theta Pinch reactor is up to 12,000 MW(th).

Considering that fusion reactor designs are still very preliminary, it is not worthwhile to describe them in the detail given in Section IV.1 of this chapter. Certainly there is more detail than given here, and the reader is referred to a recent publication [IV-30], which includes more of the plasma-physics, thermal, and mechanical design details.



1. PLASMA
2. TOROIDAL FIELD COIL (12)
3. RETRACTABLE DIVERTOR COIL (8)
4. TRANSFORMER COIL (10)
5. EVACUATED PRIMARY CONTAINMENT BUILDING
6. Li PRIMARY SYSTEM (ONE FOR EACH OF 12 MODULES)
7. Na SECONDARY SYSTEM (" " " " " ")
8. DIVERTOR Li PRIMARY SYSTEM
9. DIVERTOR Na SECONDARY SYSTEM
10. TURBINE-GENERATOR BUILDING
11. HOT CELL REPAIR AREA
12. MAGNET SHIELD
13. BLANKET
14. DIVERTOR COLLECTION AREA
15. MOTORIZED MODULE SUPPORT VEHICLE
16. AUXILIARY EQUIPMENT AREA

Figure IV-14: Wisconsin TOKAMAK Fusion Reactor, UWMAK-I

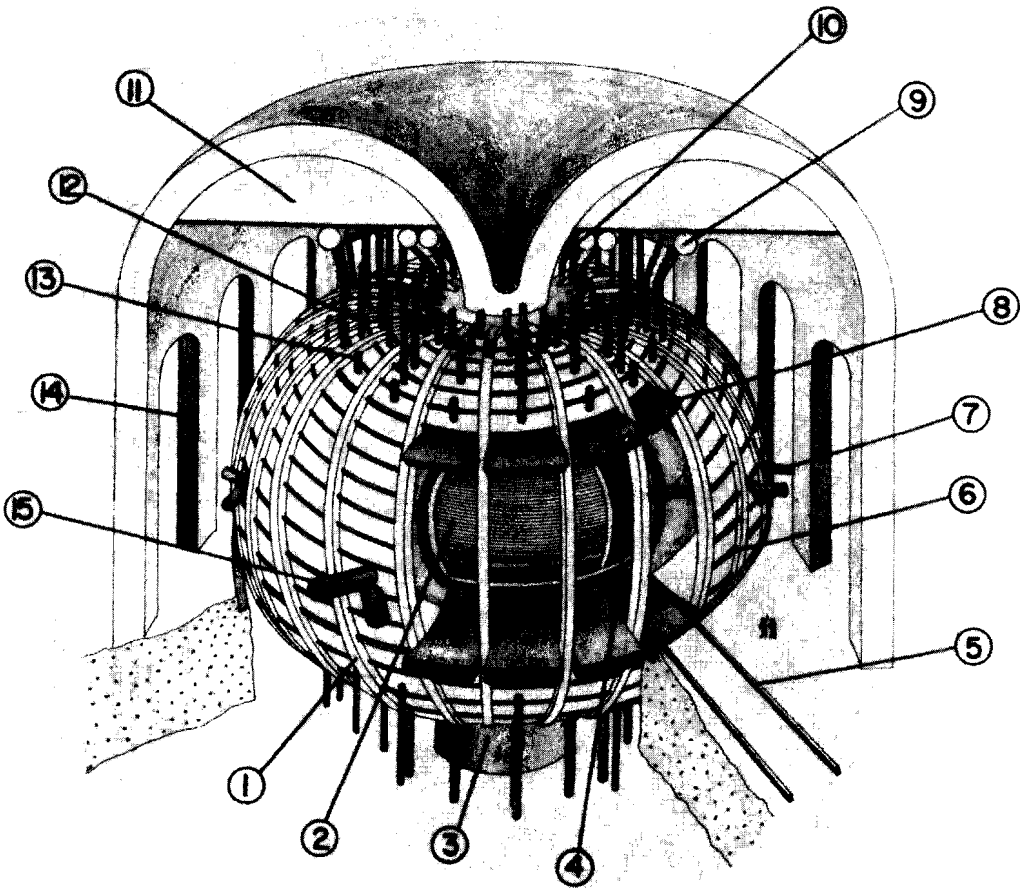


Figure IV-15: Wisconsin TOKAMAK Fusion Reactor, UWMAK-II

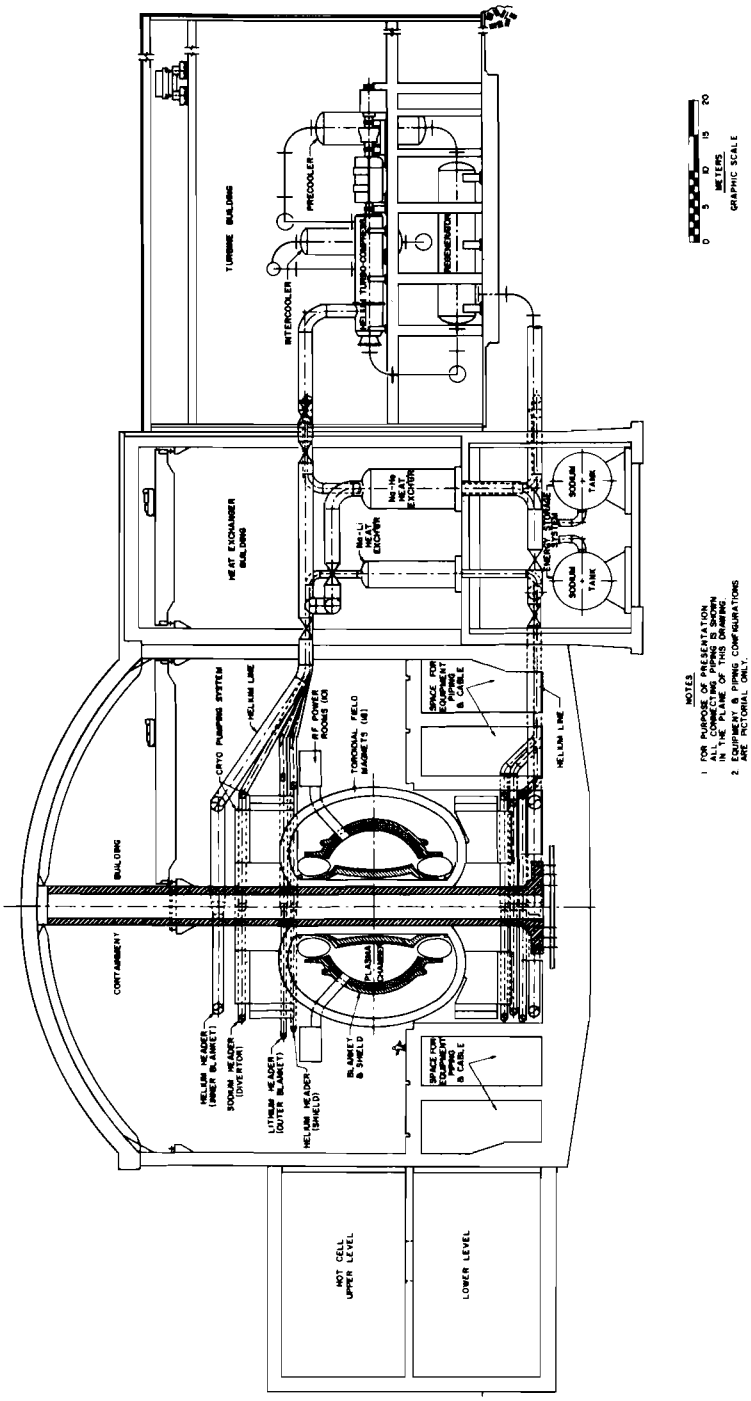


Figure IV-16: Cross-section View of TOKAMAK Fusion Power Plant UWMAK-III

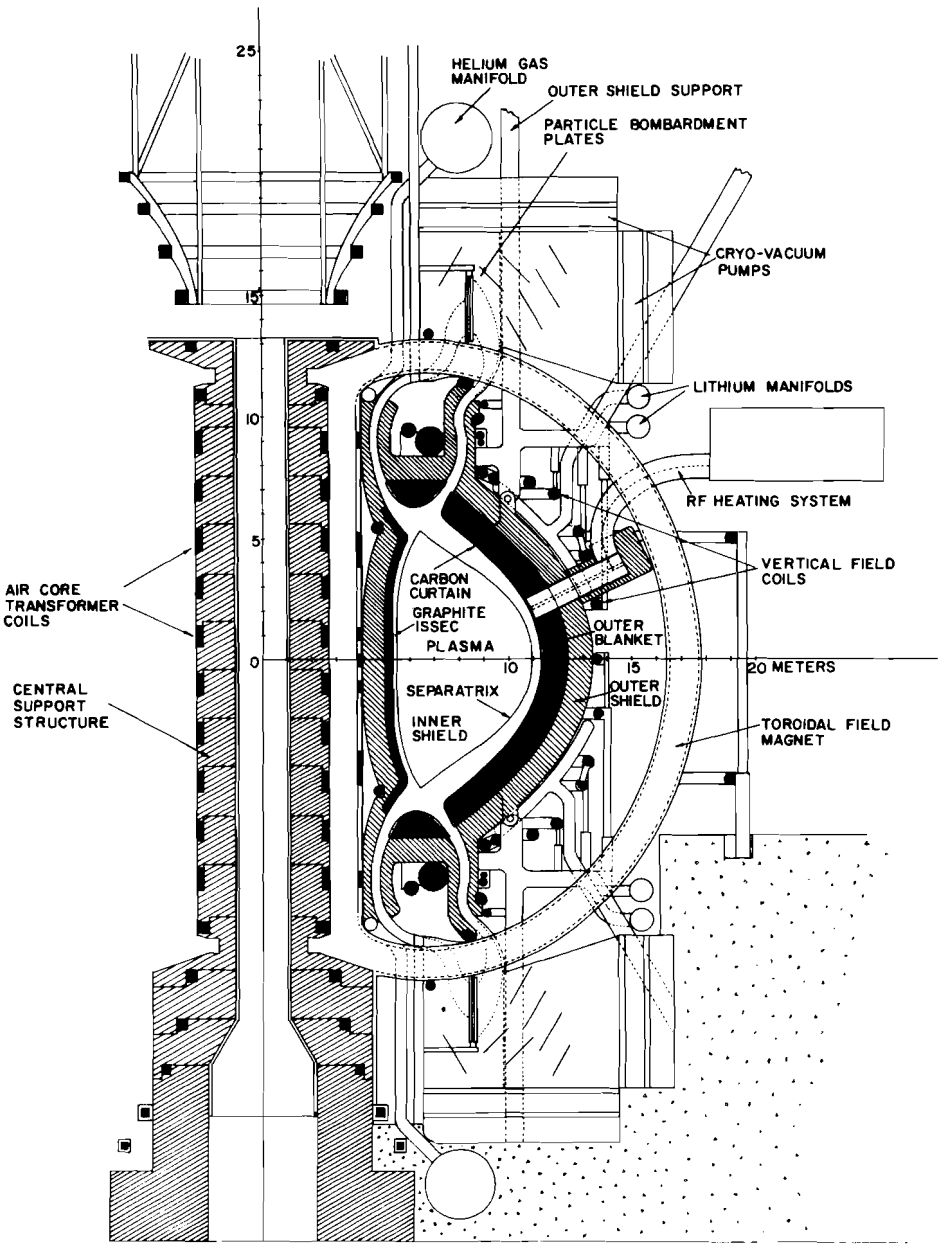


Figure IV-17: Cross-section View of the Right Half of the Nuclear Island in UWMAK-III (The system is toroidal and therefore symmetric about the vertical axis)

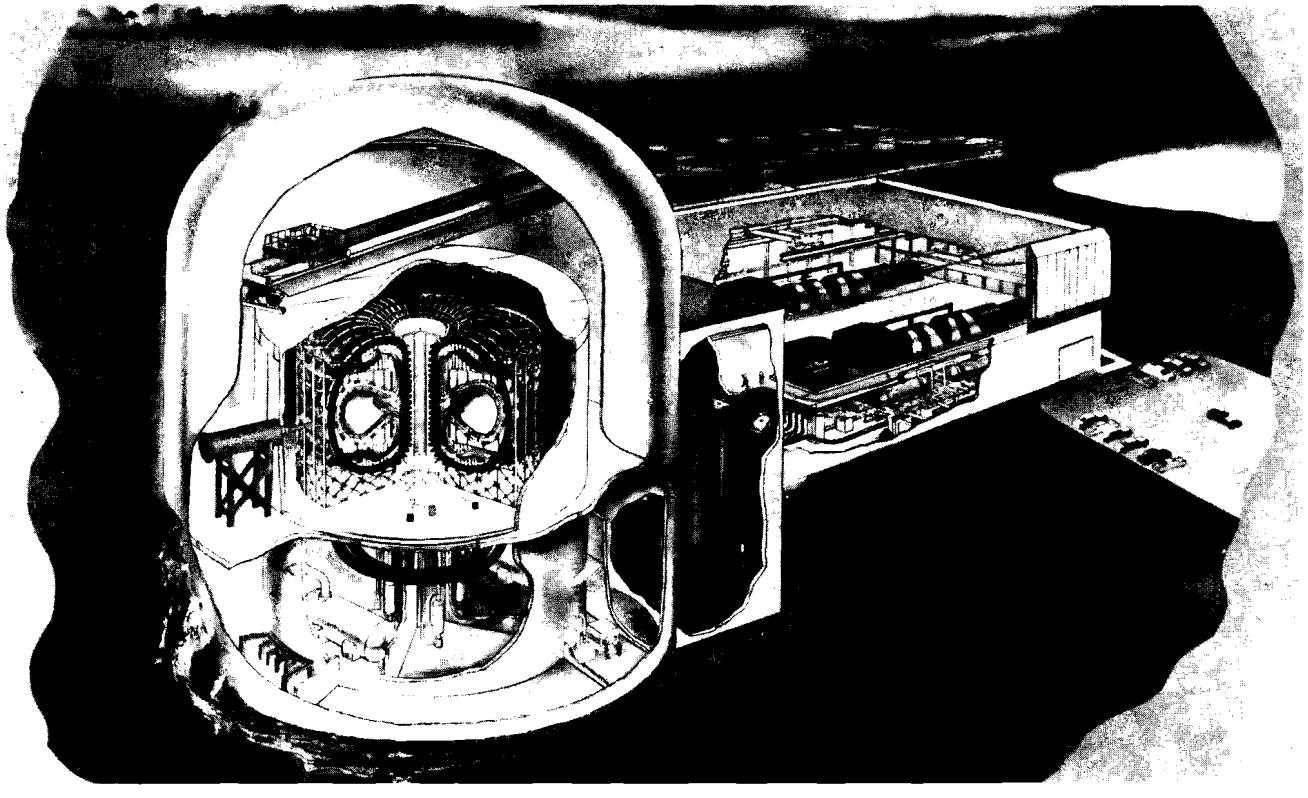


Figure IV-18: Artist's Conception of Princeton Plasma Physics Laboratory (PPPL) TOKAMAK Reactor Design

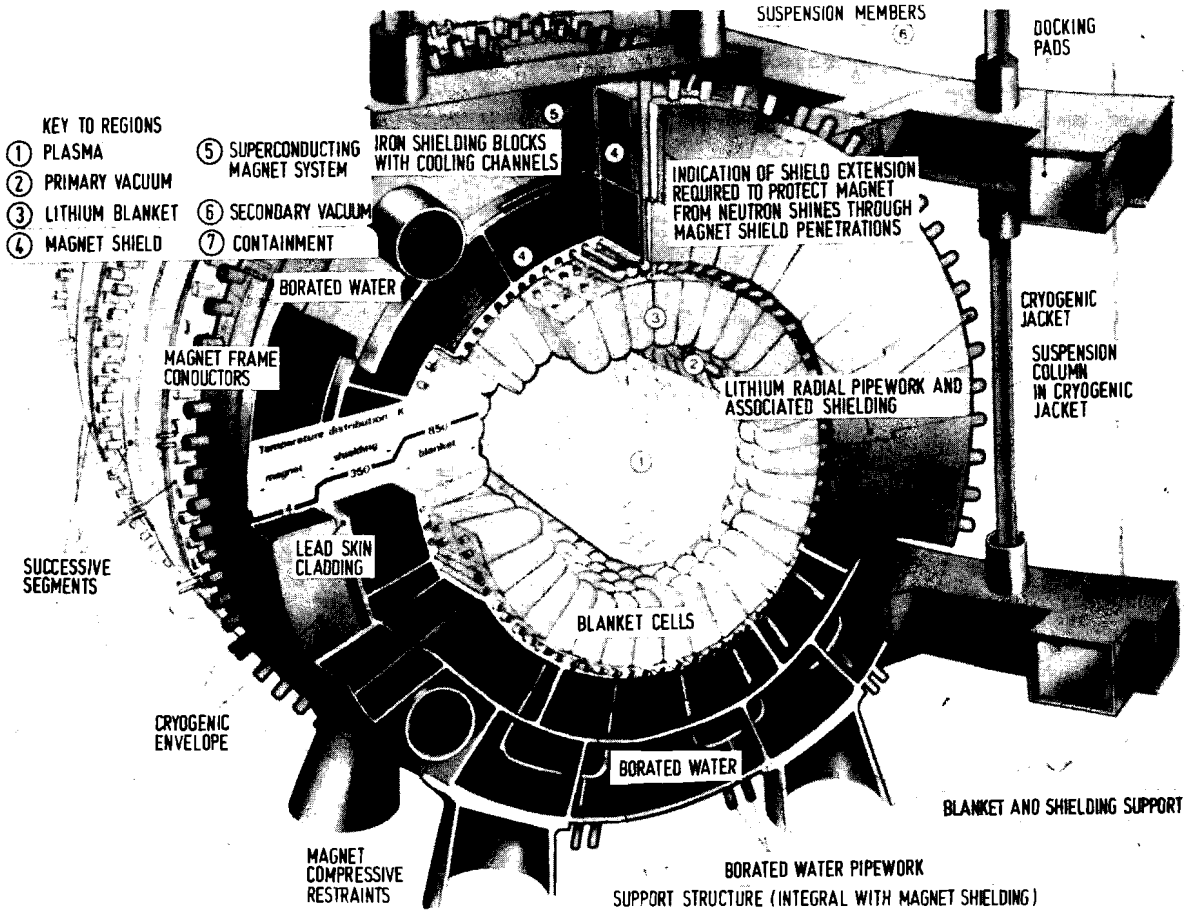


Figure IV-19: Cross-section View of Culham TOKAMAK Reactor

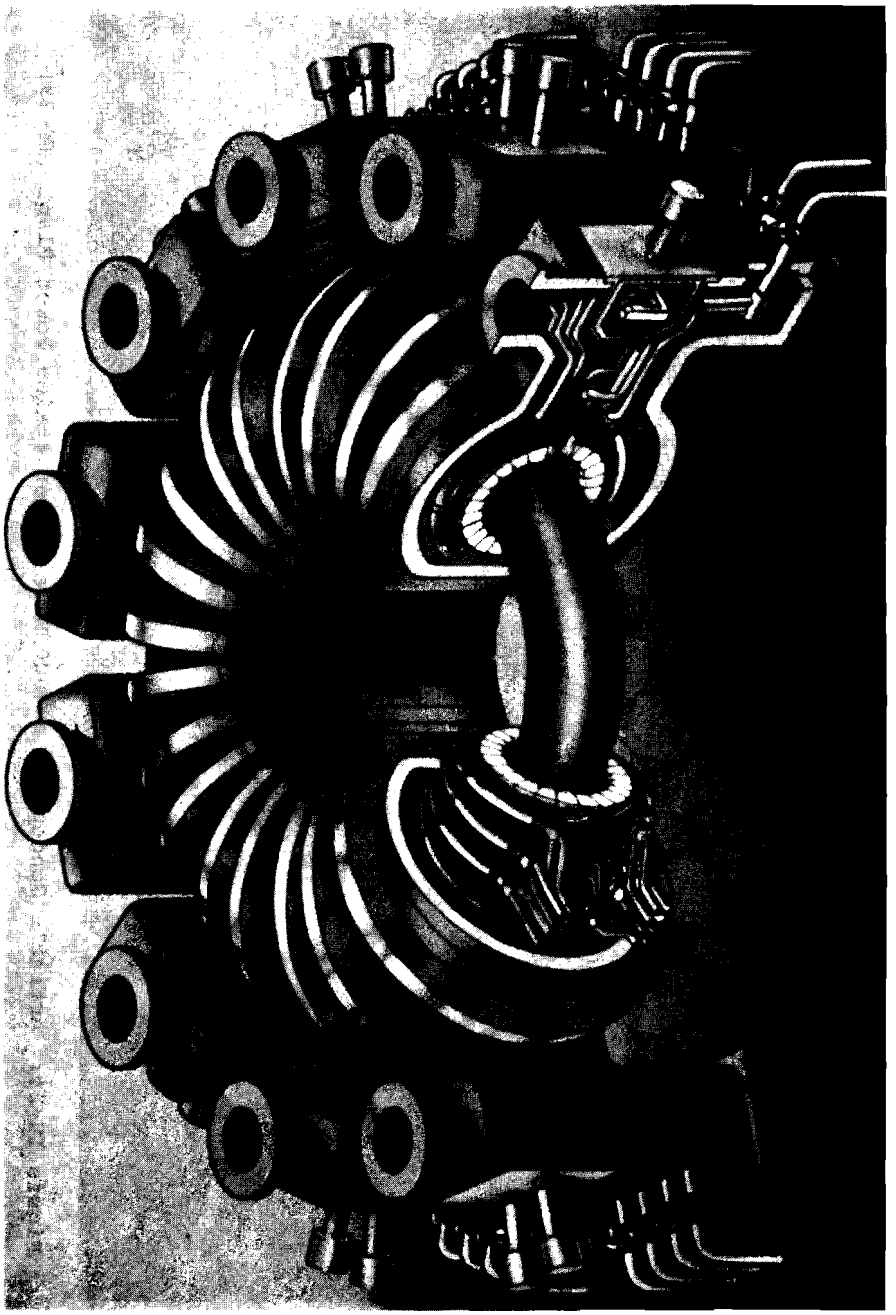


Figure IV-20: JAERI Gas Cooled TOKAMAK Reactor

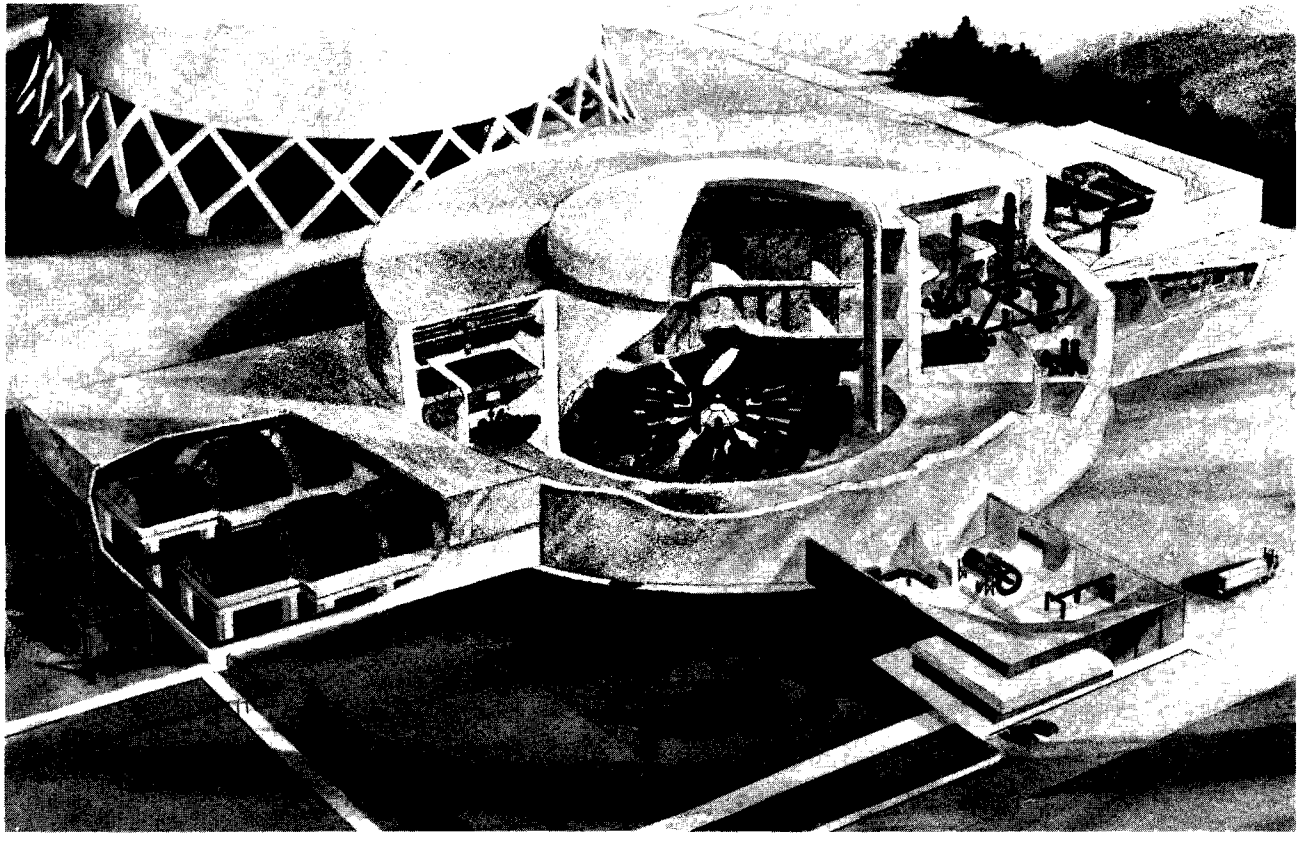


Figure IV-21: Artist's Concept of a 1000 MW(e) Laser Fusion Power Plant [IV-29]

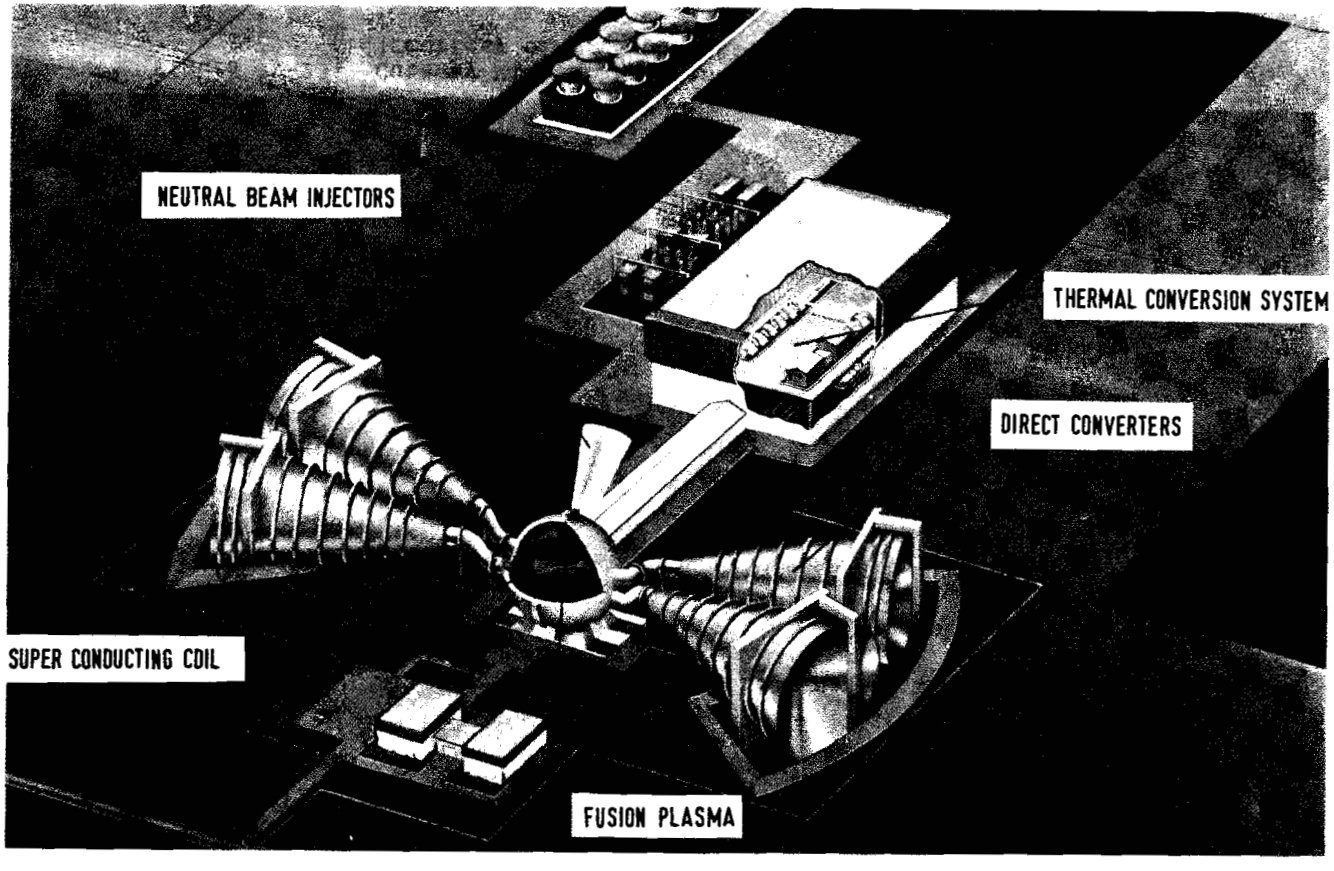


Figure IV-22: Conceptual Mirror Fusion Reactor with Direct Conversion
(Courtesy of G. Carlson, Lawrence Livermore Laboratory)

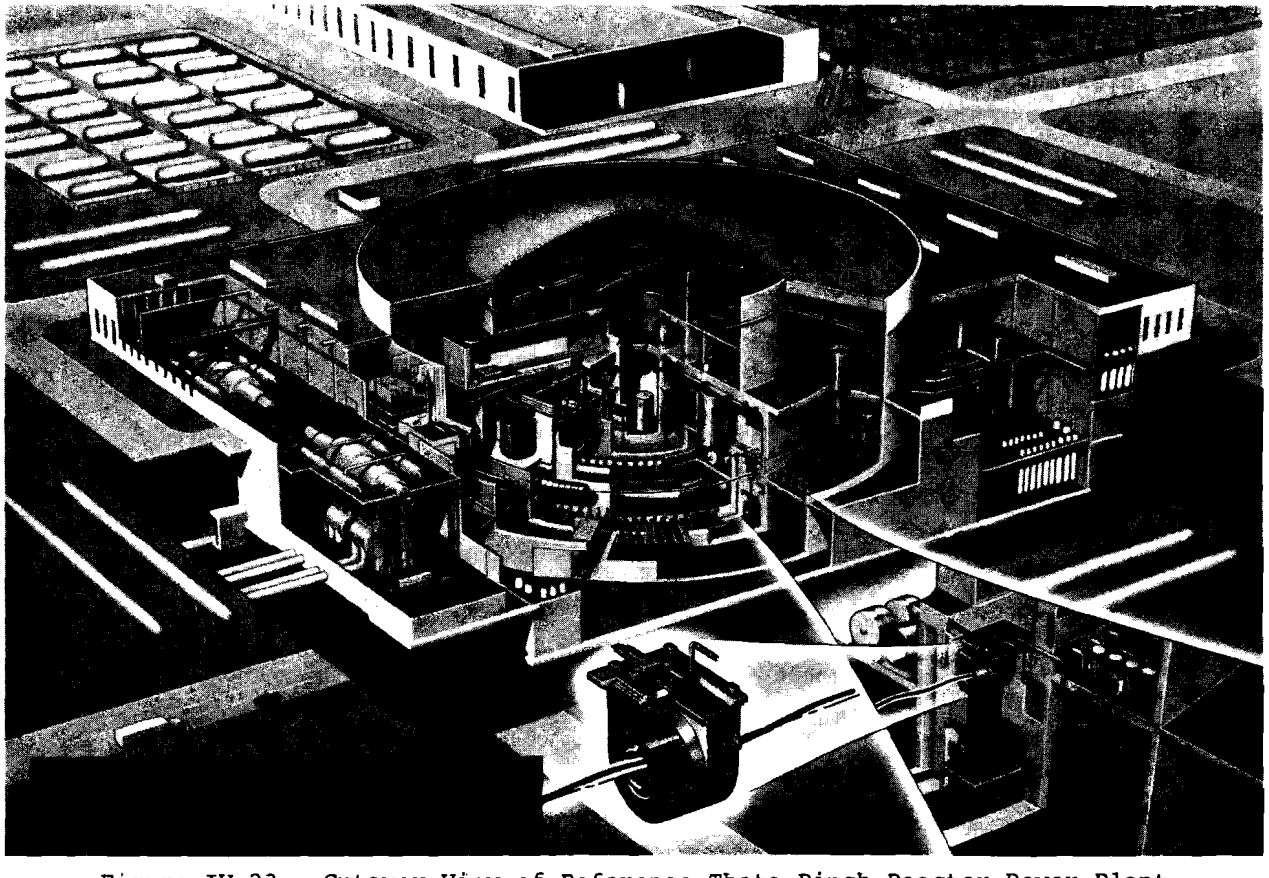


Figure IV-23: Cutaway View of Reference Theta Pinch Reactor Power Plant
(Courtesy of R. Krakowski, Los Alamos Scientific Laboratory)

3. CONCLUSIONS

A useful comparison of fusion and fission breeders requires that the analysis be undertaken at a level of detail that can only be provided by reference to specific reactor designs.

The basis for our choice in fission is that the Liquid Metal Fast Breeder Reactor (LMFBR) clearly dominates research and development programs on breeder reactors around the world, making the LMFBR by far the most likely breeder for commercialization. Historically, fast reactors have been preferred to thermal breeders because of their higher breeding ratios, which provide optimum fuel utilization and the possibility of a relatively rapid expansion of the number of reactors. Among fast reactors, liquid-metal-cooled reactors have received much more attention than gas-cooled reactors for partly technical and partly historical reasons. In any case, no prototype gas-cooled fast reactors are under construction at present. Among various existing LMFBR designs we chose the German/Belgian/Dutch fast breeder prototype reactor SNR 300 because we had full access to all the details of the program.

On the fusion side, it is impossible to state with any certainty which configuration will actually lead to a working reactor. At the present time, the TOKAMAK concept seems to provide the greatest promise of success from a scientific standpoint and, therefore, has been the object of most conceptual reactor designs. However, its toroidal geometry and complex magnet configuration make the TOKAMAK a very difficult system to design for electricity production; the construction of such large-scale power plants will undoubtedly be more difficult than that of similar-sized fission breeder reactors. Perhaps some other approach to fusion (Mirrors, laser fusion, etc.) will lead more easily to a reactor than the TOKAMAK concept, but it is too early to say. We have chosen to discuss here the liquid-lithium cooled TOKAMAK because more extensive and detailed information has been accessible for this concept than seems to be available for other approaches.

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APPENDIX TO CHAPTER IV

A MORE DETAILED ANALYSIS OF FUTURISTIC FUSION REACTOR CONCEPTS

IV-A A TOKAMAK REACTOR AND AN MHD ENERGY CONVERSION SYSTEM (THE TVE-2500 PROJECT)

(D.K. Kurbatov, G.E. Shatalov, N.N. Vasiliev)

In cooperation with the Kurchatov Atomic Power Institute, the Institute of High Temperature of the Academy of Sciences of the USSR is carrying out design studies for a fusion power plant equipped with a TOKAMAK reactor [IV-A-1].

The main characteristic features of the project under consideration are gas cooling at coolant temperatures up to 1800 to 2100 K, and incorporation of the solid-graphite ceramic blanket. With such high working-medium temperatures MHD generators can be used; these generators have high thermodynamic efficiencies and are costly compared to traditional power plants. Besides, the realization of the high-temperature gas-cooling process suggests using the plant as a high-temperature heat source in industry (in metallurgy, chemical hydrogen production, etc.).

The principal technological scheme of the fusion power plant is shown in Figure IV-A-1. Neon was chosen as a gas coolant in the solid ceramic blanket, and is used as a working medium in the heat power cycle of the plant. After power production in the MHD generator, where about one third of the initial enthalpy is converted into electricity, the working medium successively passes through the following thermo-mechanical components: steam generator, recuperative heat exchanger, cooler, compressor, and again heat exchanger; then the neon is forced back to the reactor. The compressor is driven by the turbine supplied with steam from the steam generator.

The basic parameters of the fusion power plant are given in Table IV-A-I. At an outlet temperature of Ne of 2100 K a plant efficiency of roughly 50 per cent may be reached.

The parameters for the major components of the energy conversion system were carefully selected; the calculation results were examined in detail with regard to fission [IV-A-2] as well as fusion reactor systems [IV-A-3, IV-A-4].

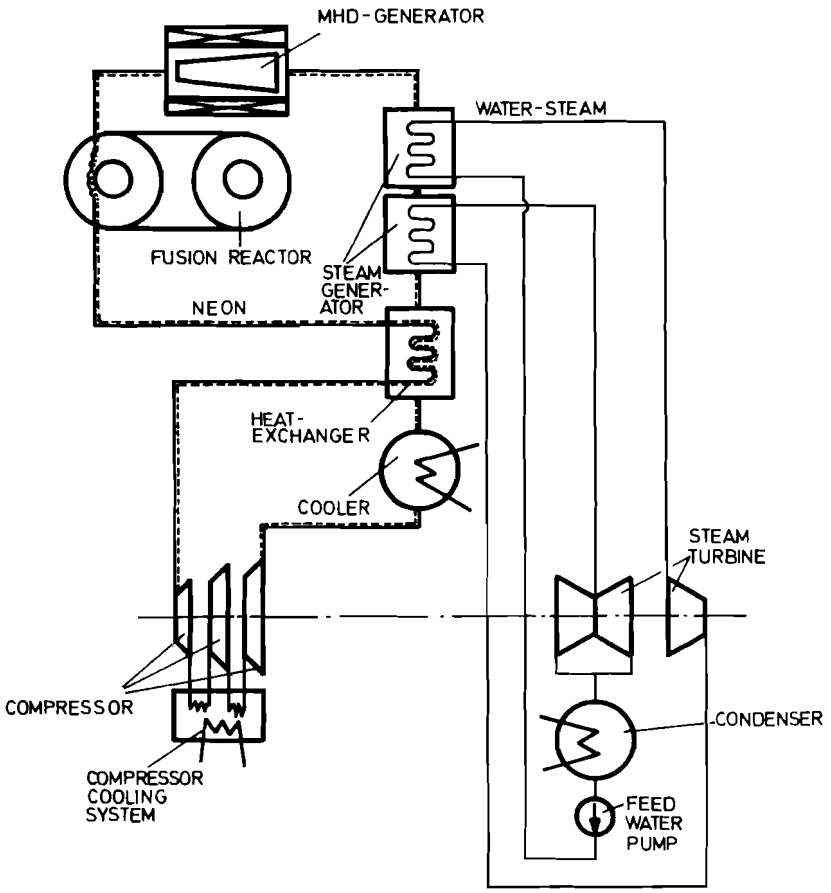


Figure IV-A-1: Principal Diagram of the Fusion Power Plant with MHD Energy Conversion System

It is, therefore, reasonable to analyze at this point the principal technical and economic parameters of the reactor unit; the way in which they were obtained is discussed in Appendix C-III. There the basic technical decisions are described that were taken when the TVE-2500 project was being worked out.

The TOKAMAK-type reactor operating on D-T fuel has been considered. The unit energy load was accepted to be 2 W/m^2; the induction of the toroidal magnetic field of the chamber axis was established to be 5 T. Before the ignition of the reaction is initiated the plasma is heated by injection of neutral deuterium atoms.

Table IV-A-I: Basic Parameters of Fusion Power Plants with a TOKAMAK Reactor and an MHD Energy Conversion System (TVE-2500)

		TVE-2500
Plant Capacity	MW(e)	2570
Efficiency	%	51
Equipment Consisting of		
TOKAMAK fusion reactor		1
MHD generator		2
Compressor with drive (steam turbine plant)		2
Thermomechanical block		2
Maximum Cycle Temperature	K	2100
Minimum Cycle Temperature	K	320
Maximum Pressure in Loop	atm	7.5
Magnetic Systems of Reactor and MHD Generators		Superconducting
Parameters of Magnetic Systems		
Induction of toroidal magnetic field at magnetic axis of reactor chamber	T	5
Maximum induction value of magnetic field at windings of reactor magnet	T	12
Induction value of magnetic field in the channel of the MHD generator	T	8
Material of superconducting busbar		Nb_3Sn+Cu
Total weight of magnetic windings including weight of stabilizing and reinforcing materials	t	1200
Tritium Inventory	kg	2

The reactor-plasma parameters were determined via a system of one-dimensional non-steady state equations which describe the balance of energy and particles [IV-A-5]. In the calculations, neoclassical expressions were used for ion heat conductivity and diffusion, and pseudoclassical expressions for the electronic component. In addition, a number of corrections concerning the anomaly of electronic-heat conductivity and plasma conductivity were inserted.

In accordance with plasma-physics calculations, the required outlet power for the injectors proved to be 100 MW, the time for reaching nominal reactor power being 10 to 15 sec. The principal criteria for the blanket design were to obtain a high-temperature working medium and acceptable tritium-breeding ratios.

The first requirement simply stipulates the usage of a solid ceramic blanket. Today the best heat-proof material containing lithium is known to be lithium aluminate: $Li_2O(Al_2O_3)$ - $Li_2O(Al_2O_3)_2$. The melting temperature of this composition depends upon the quantity of the poor lithium content phase $Li_2O(Al_2O_3)_2$, and in no case exceeds 2000 K. The working temperature of lithium aluminate will not exceed 1600 to 1800 K. This means that the blanket should be designed in a way as to provide maximum heat release, not in the lithium containing zones, but in the regions made of more heat-resistant materials such as graphite.

The neutron-physics calculations of the reactor blanket were carried out on a BESM-6 computer with the one-dimensional BLANK program using the Monte Carlo method in an energy field of 0.01 MeV to 14.2 MeV, and the P_1-approach for energy of 0 to 0.01 MeV [IV-A-6]. The calculations were carried out for plane geometry, which means that one is not confined to specific system dimensions. The statistical precision of the calculation of basic parameters was two to three per cent.

According to preliminary results, it has become clear that in order to obtain a tritium-breeding ratio exceeding one, it is necessary to use neutron multiplying material and to have lithium enrichment with Li6 isotopes. Analoguous results were received in studies [IV-A-7, IV-A-8] where lithium enrichment with Li6 was accepted to be 90 per cent, and the blanket included beryllium zones.

However, the use of beryllium, or its compounds such as beryllium oxide, or the intermetallide Be_4Zr leads to a considerable rise in costs of the blanket and the plant as a whole. In addition, the low radiation resistance of Be will make frequent replacement of the respective blanket zones necessary, which substantially increases the variable calculated cost component. The shortage of natural beryllium resources may also turn out to be a decisive factor.

Another possibility is to use a heavy-element (n,2n) reaction. Zirconium probably offers the most favorable combination of nuclear and thermal properties, and it was accepted as multiplying material for reactor blankets of TVE-2500 plants.

It was decided that in the Project an Nb-Zr alloy be used as first-wall construction material. In order to reduce the influence of a wall sputtering on the reactor plasma parameters, covers consisting of silicon or titanium carbides were used.

Tritium production is carried out in the zones which, from a construction point of view, were made according to the principle of the "ventilated fuel element": hardened or freely dissipated microsheres of lithium aluminate ($Li_2O(Al_2O_3)$) are locked up in tightly closed ampules made of Nb-Zr alloy or Mo alloy. Within the ampule, a weak flow of the inert gas-carrier is produced (for TVE-2500 helium was chosen), which carried the tritium diffusing from the microelements right to the catching system. An analoguous system of tritium extraction is suggested in studies [IV-A-8].

The zones in which the multiplication and moderation of neutrons occur can be seen as a form of a free spherical accumulation.

Zirconium may be used either as a metal, or as a compound with a high melting point, for example, in the form of the oxide ZrO_2 (with a melting point of 2800°C), nitride ZrN (3100°C), or carbide ZrC (3400°C); however, the absorption and moderation of neutrons in oxygen, nitrogen, or carbon atoms leads to a reduction of the tritium-breeding ratio as compared to the case when metallic zirconium is used, which has a melting point of 1800°C.

In the TVE-2500 project zirconium carbide, ZrC, is used for the zones in which neutrons are multiplied.

The calculation scheme of the blanket is shown in Figure IV-A-1. Neutron-physical calculations showed that with metallic zirconium the tritium breeding ratio is $BR_t = 1.09$. If zirconium carbide is used, BR_t is reduced to 0.96 to 1.01.

Thus, for a TOKAMAK reactor with a high-temperature blanket, an outside tritium generator may prove necessary. Such a generator may, for example, be a fission reactor, a fission-hybrid reactor, or a "clean" reactor with a tritium-breeding ratio that considerably exceeds one. One generating reactor may provide a group of 5 to 10 power reactors with tritium [IV-A-4] This will not be necessary if metallic zirconium is used; but then the maximum temperature of the heat carrier will not be more than approximately 1800 K for the cooling system examined here.

The toroidal magnetic system of the TVE-2500 reactor consists of 36 D-form coils locked in cryostats and fixed to a supporting steel cylinder, a protection system, electrical engineering equipment, and cryogenic equipment.

The toroidal magnetic field at the chamber axis is accepted to be 5 T. In this case, the maximum induction of the field at the windings is 12 T. This led to the choice of copper-stabilized Nb_3Sn as superconducting busbars. At the present time, technology for manufacturing superconducting busbars on the basis of Nb_3Sn is less developed than that of busbars or wires on the basis of Ti-Nb or Nb-Zr alloys. But this may change before long.

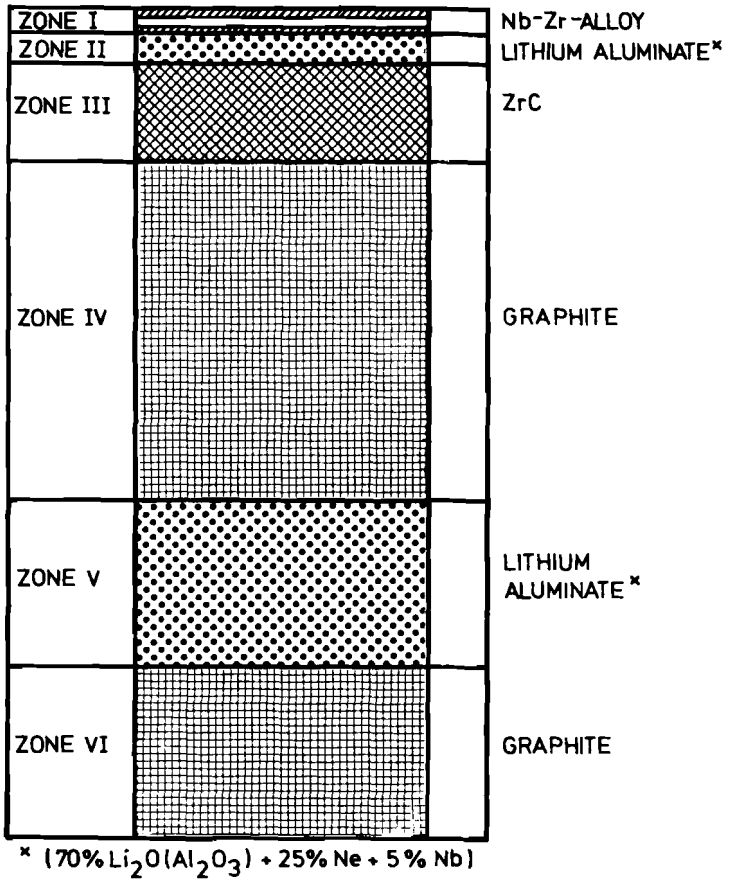


Figure IV-A-2: Calculation Scheme of the Hexazone Blanket

At the same time, the application of strong magnetic fields makes it possible to increase the specific output of the plant ($\sim B^4$), so that, on a long-term basis, niobium stanide will be used in the future. In plants of the TVE-2500 type, the use of niobium stanide is also justified because strong magnetic fields are required for MHD-generators.

The working medium was chosen with a view to simple plant operation, efficient blanket cooling, and achievement of acceptable electrophysical properties of the plasma in the channel of the MHD-generator. These requirements are best met by neon, to which, for an efficient operation of MHD-generators, a minor quantitiy of an easily ionizing seed must be added (0.05 per cent per volume of cesium).

The conversion of thermal energy into electrical energy of the heat carrier takes place in the MHD-generator channel. The temperature and pressure levels at the entrance to the MHD-generator (2100 K, 6 atm) are such that the equilibrium thermal ionization of cesium does not suffice to create an acceptable electric conductivity of the working fluid plasma. Therefore, the parameters of the working process in the channel were chosen to guarantee a sufficiently high non-equilibrium ionization. In carrying out the plasma-dynamic calculations of the MHD-generator channel, the critical (efficient) Hall number was accepted to be $\beta_* = 1$.

With due regard for the results of these calculations and numerous experimental data collected in model plants, a linear MHD-generator of the Faraday type with sectional electrode walls was chosen. It was proposed to use as electrode material pyrographite, or high-temperature compositions on the basis of zirconium oxides and rare earth elements. The magnetic field in the MHD-generator channel was accepted to be 8 T with a maximum induction at the windings of 11 T.

Analyses of different compressor drives (electro-motors, steam or gas turbines) have shown that the most economical and simplest method is to use a steam turbine drive. In this case, standard equipment with an extraordinarily big capacity per aggregate may be used. In the TVE-2500 project, two standard steam-turbine plants with supercritical steam parameters (565^{0}C, 240 atm) are used with a capacity of 500 MW(e) each. The plants also include water preparation systems, water systems, condensers, heat exchangers, etc. The steam for the turbines is generated in the steam generator (see Figure IV-A-1].

The heat exchange equipment of the working-fluid loop (steam generators, heat exchangers, etc.) is planned according to "traditional schemes", where the parameters of the thermal cycle were selected in such a way as to guarantee the operation at temperatures permitting the use of heat resistant and stainless steels.

The above mentioned features of the most important aggregates of the plant--reactor, MHD-generators, and thermomechanical block--led to the selection of a specific scheme presented in Figure IV-A-3.

After estimating the optimal unit power rates of the major scheme components, it was concluded that the fusion power plant with a 5000 MW(th) reactor should consist of two self-contained MHD units, each rated at 1250 MW of installed electric capacity. In principle, only one power unit may be used, but this would require complex auxiliary systems.

The reactor unit, MHD-generators and heat-transfer and mechanical equipment, including a compressor, are situated in the main power plant building. The latter also contains special boxes (hot cells) for separate reactor module repair. The

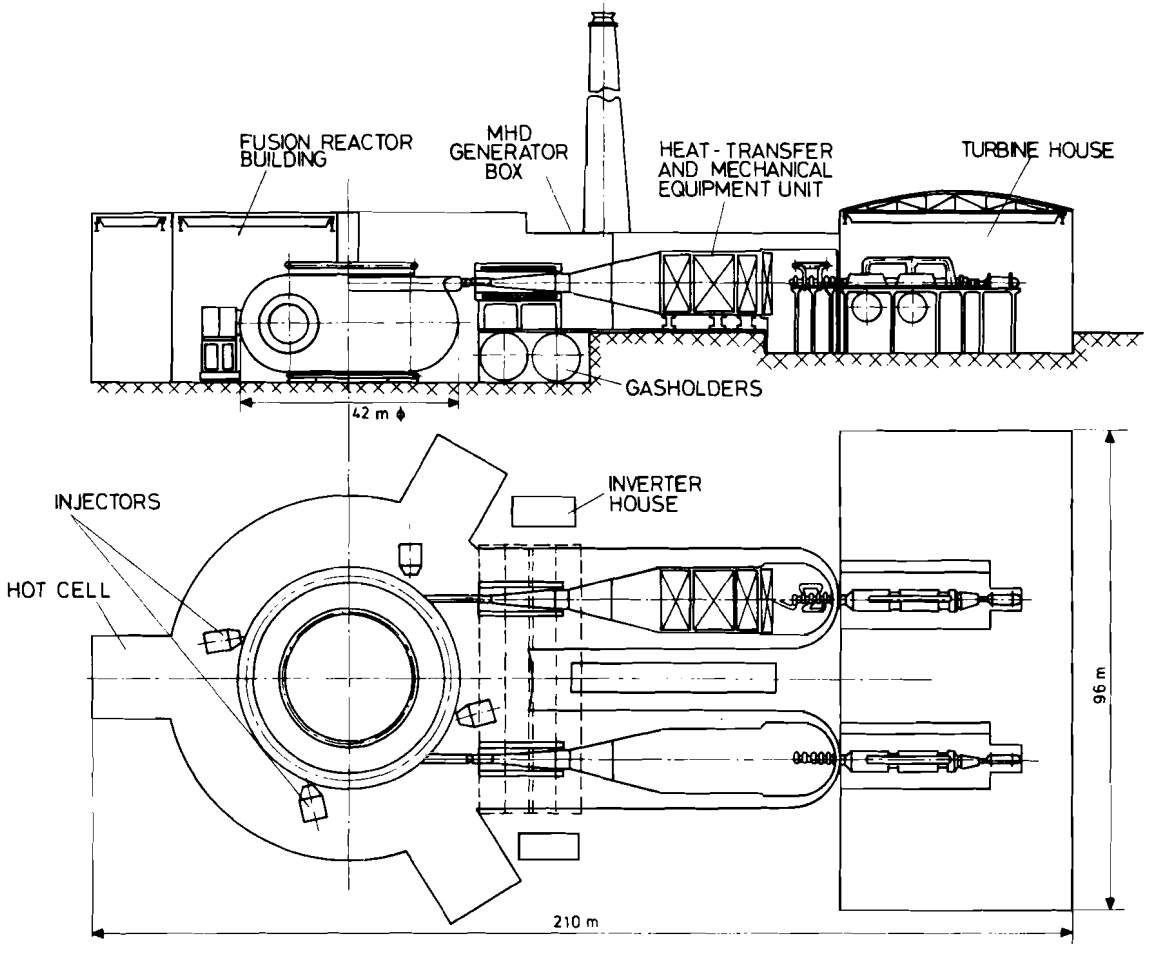


Figure IV-A-3: Views of the TVE-2500 Fusion Power Plant

drives for the compressive steam turbines are situated in a separate building. The reactor building is made of concrete and is partly deepened under the ground.

The electrical current network as well as hot and cold piping of the coolant system are arranged in the MHD-generator boxes. These boxes and the heat-transfer-equipment pits are enclosed in a hermetic containment to avoid the consequences of possible destruction of the MHD-structures and/or break-down of the gas pipelines. The compressor and steam turbine have a single rigid shaft; the penetration of the shaft through the wall is specially sealed.

All the boxes of the main power plant building are hermetic and connected by way of blow-up valves to the gasholders, having a volume large enough to receive the total mass of the working medium in case of its blow-down from the gasdynamic loop.

A few words should be said about the nuclear safety of the fusion power plant. The theoretical calculations of the radio-activity originating from induced nuclear reactions in the re-actor materials, qualitatively agree with the results of the UWMAK project. But the use of zirconium-based alloys as structure materials results in a lower initial radioactivity, 0.1 Ci/W(th), and leads to a decay of activity that is faster compared to the case where niobium or stainless steel is used.

The total stock of tritium at the plant is not large and equal to 2 kg, with the major portion of tritium being kept in the fuel storage and not in the blanket.

The most complicated problem is that of retaining the radioactive gaseous products. Though the coolant (neon) practi-cally is a non-activated gas, one should bear in mind that a certain amount of cesium (0.05% per volume) must be injected into the loop, in order to provide effective operation of the MHD-generator.

As a result of the nuclear reactions Cs133 (n,γ) Cs134 and Cs133 (n,2n) Cs132, two radioactive isotopes Cs134 and Cs132 are produced; they have half-lives of 2.1 years and 6.58 days, respectively. The coolant-specific activity con-centration defined by the radioactive decay of these isotopes equals 0.03 Ci/cm^3. Thus, it will be more probable that depressurization of the loop is maintained. However, on the other hand, the presence of cesium in the working medium results in a reaction of cesium with tritium leaked into the gasdynamic loop. As a result, cesium tritide (CsT) is being formed and then extracted from the loop, together with the remaining cesium at the seed-capturing and recovery system.

In addition, the safety problem during power plant accidents must also be studied.

IV-B THE LASER FUSION REACTOR DESIGN LTB-500

(R.R. Grigoriants)

The energy block project with the laser fusion reactor LTB-500 was carried out together with the Institute of High Temperature of the Academy of Science of the USSR and the PN Lebedev Physics Institute, Academy of Science of the USSR.

The basic results of the studies were announced at a session of a working group of specialists from the USSR, the US, the FRG, and IIASA [IV-B-9]. This session was held in Moscow in May 1976.

The aim of the LTB-500 project is to determine the preliminary technical and economic indices of a fusion power station with laser initiation. Furthermore, this project made it possible to formulate more clearly the basic requirements of construction and system elements.

Table IV-B-II puts together some basic parameters of the LTB-500 laser energy block.

Table IV-B-II: Basic Parameters of the Laser Energy Block LTB-500

		LTB-500
Basic Equipment - Thermonuclear reactor with laser initiation, wetted wall, condensing turbine K-300-240, electric discharge CO_2 laser		
Laser Radiation Energy Supplying the Target	J	$5 \cdot 10^5$
Coefficient of Tritium Reproduction		1.25
Efficiency of Laser System		0.05
Coefficient of Laser Heat Utilization		1.0
Pulse Frequency		10
Heat Output of the Reactor	MW(th)	522
Electric Output	MW(e)	170
Coefficient of Efficiency of Energy Block		0.33
Specific Capital Investments	Rub/kW(e)	370
Cost of Electric Energy	kopeck/kWh(e)	0.75

1. HEATING SCHEME

The energy block LTB-500 includes the following main aggregates:

- a "clean" fusion reactor with laser initiation;
- the condensing turbine K-300-240;
- electric-discharge CO_2 laser.

The heating scheme of the plant is based on the serially produced K-300-240 turbine. When entering the turbine the steam has a temperature of 565°C, and a pressure of 240 atm. The turbine capacity of the generator busbars is 300 MW(e). The turbogenerator requires intermediate superheating of the steam at a pressure of about 40 atm. The operation of the turbine is guaranteed by a system of regenerative heaters, a deaerator, condenser, condensate, feedwater pumps, and other auxiliary equipment. In order to provide for regenerative heating and steam intake in the turbine deaerator, a number of steam outlets are envisaged. Through one of them the turbo-drive of the feedwater pump is provided with steam. The total heat consumption per turbine, including the turbodrive for the feedwater pump, amounts to 1850 kcal for 1 kWh(e).

The heat produced in the reactor is removed by lithium, and reaches the water circulation through a system of heat exchangers (water heaters and steam generators).

The laser-fusion reactor, which serves as a heat source, has a chamber wall that is protected by a lithium layer reaching to the surface of the chamber (reactor with "wetted" or "sweating" wall), and a lithium blanket.

Lithium was selected as the heat carrier first of all because of tritium reproduction. The temperature of lithium, which is 700°C when leaving the reactor, guarantees the long stability of the nickel-free steel in a lithium environment.

Taking into account that there is a loss of heat in the pipes, the temperature of steam when leaving the steam generator must be 580°C. The excellent thermal properties of liquid metals, even with a comparatively small decrease of the maximum temperatures of lithium and steam, made it possible to introduce an additional intermediate loop between the first, i.e. lithium, and water loops. The heat carrier of this loop is the eutectic alloy Na-K, which has a low melting temperature of -12°C, so that no complications arise in connection with the heating of pipes of the intermediate loop. The introduction of this loop eliminates the penetration of tritium into the steam circuit. Furthermore, this makes it possible to avoid heat exchanging surfaces "lithium-water". The technology of the heat exchangers "(Na-K)-water" is, however, throroughly studied and mastered.

The first (i.e. lithium) loop of the energy block has two branches: one for the lithium which evaporates and which is heated within the reactor, and one for the lithium going through the blanket. The reason why this solution was accepted is the different tritium concentrations in these branches. At the bypass of these branches there are systems of tritium separation which work by the principle of tritium diffusion through separating niobium walls.

The CO_2 laser, based on a mixture of N_2-CO_2 gases, was considered in the project because it is one of the prospective types of lasers for LTB. It is assumed that the efficiency of this laser will be from five to ten per cent. For the project five per cent were accepted. With the electic-discharge CO_2 laser there is a good possibility of using part of the energy which is not converted into light in the laser. This part of energy, which can be called laser heat, is passed on in the heat exchanger to the feedwater after it has gone through the regenerative heaters. The inclusion of the heat exchanger, which contains a N_2-CO_2 mixture and water, will reduce the heat exchange surface in the low-temperature range, or slightly simplify the work of the laser system.

However, on the other hand, it will unfortunately oust heat regeneration and require that the steam extraction in the accepted standard turbine be newly computed.

The N_2-CO_2 mixture is pumped through by means of a compressor of 20 MW power.

2. FUSION REACTOR

In carrying out the project, four variants of reactor chambers were examined. They differ with regard to the protection of the first wall:

- wetted and sweating walls, respectively;
- protection of the reactor walls from the flow of charged particles by means of a magnetic field;
- use of an ablation layer;
- use of a rotating lithium funnel.

Each of these possibilities has its advantages and difficulties and requires a more detailed analysis of all problems, such as: a solution to the question how the target is to be reached by laser radiation; requirements of target input; necessary frequency of microexplosions in the chamber; reaction of the reactor chamber to shock loads; radiation and corrosion stability of the chamber construction; etc.

The wetted wall forms a protective layer of lithium along the entire inner spherical surface of the chamber. The spherical form of the first wall is advantageous from the viewpoint of an even distribution of the neutron fluence on the first-wall construction, but also with regard to the reaction to shock loads caused by microexplosions. The reactor includes the chamber, the blanket, and the technological protection system.

The first wall of the reactor chamber consists of a network. The lithium passes through the network and comes from the collector which is situated behind the first wall. The blanket is to be found behind the collector and consists of a spherical layer of lithium with a thickness of roughly 60 cm and a graphite reflector. Laser radiation is supplied to the target through 12 ducts placed at the apices of a dodecahedron. The construction material of the chamber and the blanket, which comes into contact with lithium, is nickel-free steel.

The project paid special attention to the determination of the chamber radius. Calculations were carried out which determined the mass of evaporable lithium in the chamber, the pressure and temperature of lithium in the reactor chamber, the energy of the explosion wave caused by the interaction of the "combustion" products of the pellet and the lithium layer on the chamber wall, and the pressure to be passed on to the blanket. One also calculated the time necessary for the "combustion" products of the pellet and the evaporable lithium to be blown down from the chamber. The latter quantity is extremely important, for the frequency of microexplosions of the targets depends on this quantity. The heat output of an LTB is

$$N_t = E_{las} \cdot K_{BR} \cdot f$$

where:

E_{las} = energy supplied to target with laser radiation ;

K_{BR} = coefficient of energy breeding (is chosen to equal 100) ;

f = frequency of microexplosions .

The calculation of the heating scheme makes clear that the heat output of the reactor must be about 500 MW. Thus, with $E_{las} = 5 \cdot 10^5$ to 10^6 J, which is an acceptable quantity for long-term laser systems, f must be within the limits of 10 to 5.

The calculation of all quantities enumerated above depended on the chamber radius.

The interaction between charged particles and the lithium layer was calculated according to well-known formulae for the energy losses due to the ion movement in matter. The calculated estimates of parameters were determined by two methods:

- assuming a partial-phase equilibrium between the volumes of evaporated and liquid lithium;
- having due regard to the heat radiation from the chamber.

The basic decisions on the reactor construction were made carefully and in accordance with the rather unfavorable results of one or the other calculation. The radius of the first wall is accepted to be five m. This radius makes it possible to distribute the outlets for the combustion products of the target and the evaporated lithium over an area of about four per cent of the entire area of the first wall. The time necessary for clearing the chamber will be 0.1 sec., which guarantees a frequency of 10 cycles/sec.

With the accepted radius, the shock loads acting on the chamber construction are comparable with those loads for which present day constructions are calculated.

The integral neutron fluence for the first wall of such a reactor will be $2.5 \cdot 10^{21}$ n/cm^2 in two years of operation.

The reactor size as well as the cheap materials used make it possible to apply a simple and linear method in solving the problem of replacement after the useful life of the reactor or major accidents. After remote cutting of the pipes, the reactor is removed from the box and transported to the hot chamber. Minor damage is removed with the help of manipulators or robots.

3. ARRANGEMENT OF A POWER PLANT WITH A FUSION LASER REACTOR

The basic arrangement of the main building of the power station is the block scheme. Widely used in the USSR, this scheme, which allows for expansion by means of additions of new blocks, is based on the transversal arrangement of turbo-aggregates in the machine hall. The building is planned in accordance with fire and sanitary regulations in force in the USSR.

The reactor block is located below zero elevation, which makes it possible to arrange it comfortably with the laser-system room and reduce the quantity of concrete needed. Of course, when actually selecting the area, the question of a

possible deeper placement of the reactor box must be specified, e.g. in connection with the hydrogeological characteristics of the ground.

The laser system is placed in the acceptable vicinity of the reactor box. The light guides have a number of bends for the reduction to permissible values of radiation penetrating into the laser-system room. This is a room with restricted access in periods of reactor stop, with a protective slide covering the light guides.

The block switchboard, deaerater room, etc., are placed between the reactor room and the machine hall. The tritium systems are in the insulated part of the building.

The hot chamber is at the end of the building, its product intake is carried out through the transport corridor. The hot chamber is the place where irradiated construction and aggregates are cut and pressed.

The cryogenic systems which ensure cooling of the target preparation systems, vacuum systems for pumping out the heat-carrier tracts, as well as argon and nitrogen service systems, and the firefighting system may be located both between blocks and at the end of the building, depending on the number of blocks in the power plant.

Electric distribution equipment, water supply systems, administration buildings, the inert gas reservoir and the building for the destruction of liquid metal residues are located separately from the main building.

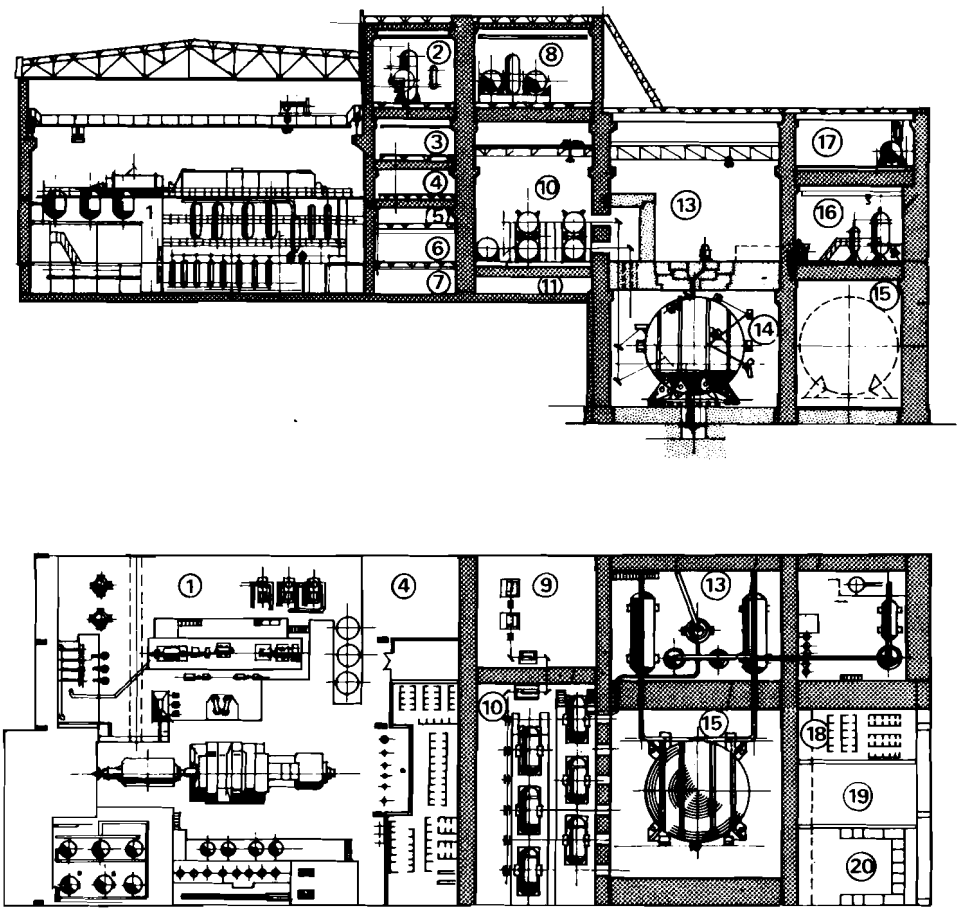
Figure IV-B-4 shows views of a power station with an LTB-500 fusion reactor.

4. SOME INDICES OF THE LASER ENERGY BLOCK LTB-500

As can be seen from the description, the values used in the project for a number of quantities were extremely moderate, e.g. the efficiency of the laser system η_{las}, and the frequency of microexplosions f.

The selection of the energy conversion system was based on well-known technologies, although for laser-fusion power stations new energy conversion schemes with higher values of efficiency [IV-B-10] and [IV-B-11] are suggested.

In the author's opinion [IV-B-9], the choice of such moderate parameters, a fairly simple and unassuming thermal diagram, as well as the simplest decisions concerning the electric power plant as a whole, must produce the necessary values of specific



1. Machine hall
2. Deaerator room
3. Pipe floor
4. Switchboard room
5. Cable floor
6. Switchboard installation room
7. Cable basement room
8. Room for water disactivation system
9. Breeding system room
10. Laser hall
11. Basement room for electric supply of lasers
12. Pipe basement room
13. Reactor hall
14. Reactor box
15. Corridor for reactor transportation
16 Room for tritium outlet system
17. Gas disactivation room
18. Switchboard for tritium outlet system
19. Dosimetry service room
20. Recreation rooms

Figure IV-B-4: Views of Power Station with LTB-500 Laser Fusion Reactor

capital investments, and an electrical energy generation cost price as near to real values as posssible. In this connection, recently started designs of laser-driven fusion reactors, D-T pellets manufacturing systems, and tritium removal systems are taken into account.

However, in spite of the moderate parameters, technical and economic indices for the block were received which, on the whole, are satisfactory. They allow conclusions with regard to the competitive value of these power stations.

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V. RADIOACTIVE INVENTORIES OF REACTOR ECONOMIES

G. Kessler, G.L. Kulcinski

1. INTRODUCTION

The comparison of radioactivity in various nuclear systems has not always been performed consistently, and has almost always been controversial. For any comparison to be perfectly correct, one would have to consider:

(a) total number of radioisotopes produced;

(b) relative toxicity of the radioisotopes;

(c) half-life of the radioisotopes;

(d) probable pathways for the release of radioisotopes into the environment;

(e) pick-up radioisotopes by humans.

This situation is graphically depicted in Figure V-1.

A compilation of the inventories of radioisotopes (in curies) for both the fission and fusion reference reactors will be given in this Chapter. This information, coupled with the maximum permissible concentration (MPC), will be used to calculate the maximum biological hazard potential (BHP) in both reference reactors. The reader should *not* draw quantitative comparisons from these numbers because not all of the radioactive inventory can be released. Chapters VI, VII, and VIII will use the information generated in this Chapter to calculate more meaningful numbers. Unfortunately, the point of ultimate importance, the amount of radioactivity actually absorbed by humans in a contaminated environment, cannot be readily calculated for all the radioisotopes in fission and fusion. Such a calculation requires a complete knowledge of residency time in the contaminated area, chemical uptake in the various food chains, intake of contaminated air, food, and water, etc. Some of this work has already been done for isotopes such as tritium, iodine, and cesium, but it has not been done for all metallic isotopes nor for all the actinides.

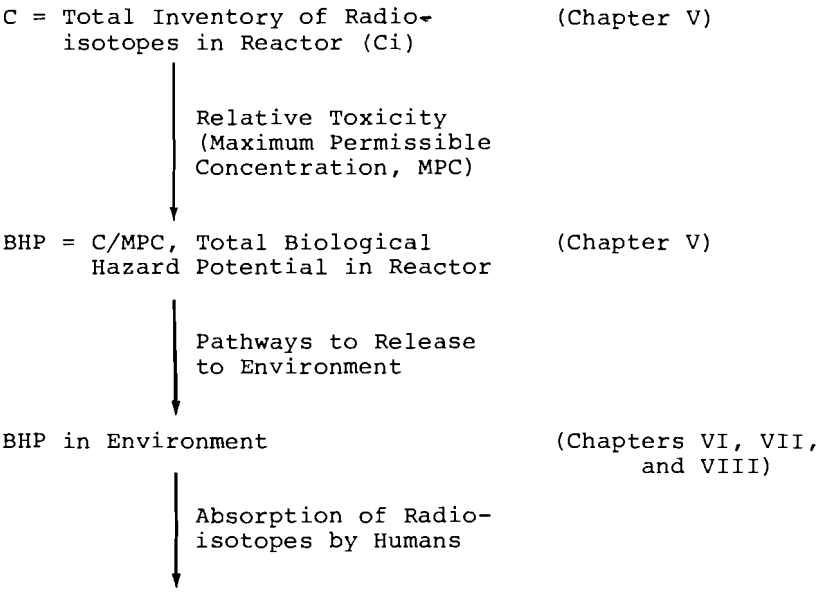

Figure V-1: Factors to Consider in the Assessment of Radioactive Inventories

The decay of the radioisotopes is important with respect to three distinct time-frames:

(1) short term, up to a few weeks, corresponding to accidental releases;

(2) intermediate term, up to a year, because of various maintenance and accessibility problems;

(3) long term, over 100 years, because of waste-storage problems.

A safety analysis corresponding to an accidental release of radioisotopes can only be performed if the pathways (Chapters VII and VIII) are clearly defined, and the probability of release by those pathways is given. This is possible in *detail* only with systems which are fully designed down to the last "nut and bolt", or after such a system has been operating. Obviously, the LMFBR has just entered this stage, while the fusion systems are still in the conceptual stage.

Much the same could be said about the maintenance aspects of nuclear systems, but the chemical form and transport of isotopes within the reactor during normal operation are now of much greater importance.

The situation with respect to long-term waste storage also depends on the release mechanism, but in a much different context. The pathway to release is no longer connected to the normal or abnormal operation of the reactor. More "conventional" release modes can be envisioned, such as corrosion of metallic or ceramic encasements, and the subsequent leaching of radioisotopes from the solidified wastes by ground or surface water. While such a calculation is in theory not difficult to make, we have not attempted it in the scope of this report.

To summarize, it is the object of this Chapter to calculate the inventory of radioisotopes in curies, and to convert that into a BHP index for both the fission and fusion reference reactor designs. Chapters VII and VIII will use this information to establish the potential BHP outside the reactor, and comparisons between fission and fusion will be made on this basis.

2. RADIOACTIVE INVENTORIES OF SODIUM-COOLED FAST BREEDERS

The consideration of radioactivity in LMFBRs is complicated by the fact that a large amount of activity is actually removed from the reactor, and is in various stages of circulation at any given time. This is in contrast to the case of fusion where, theoretically, there need be no external circulation of radioactive substances*. A simplified schematic representation of the case for fission is shown in Figure V-2, where it can be seen that, excluding the short transportation time of radioisotopes from one facility to another, there are four major distinct locations which have to be considered:

(a) fuel fabrication facility;

(b) reactor plant;

(c) reprocessing facility;

(d) waste storage.

Since we have not developed the scenario for deployment of LMFBRs, we will not consider the details of the waste-storage facilities but only normalize the input to such a facility per

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* However, since the blankets are not expected to last for the lifetime of the reactor, there will be radioactive structural material exterior to the reactor which must be processed and stored.

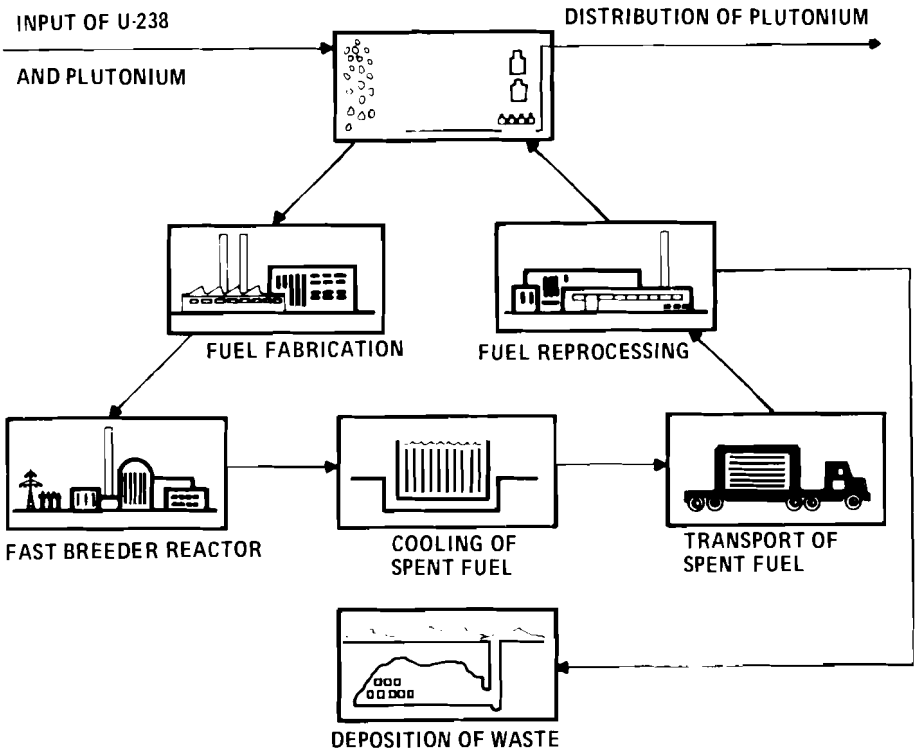

Figure V-2: Closed Fuel Cycle for Fast Breeders

GW(e)·yr. The major thrust of this Chapter will be associated with the radioactive inventory in the reactor itself, and a more cursory treatment will be given to the fuel-fabrication and reprocessing facilities.

2.1 Radioactive Inventories of the Reference Breeder Reactor Plant

There are basically five classes of radioactivity to consider in LMFBRs:

(a) structural components such as cladding, core constraints, etc.; the present calculations use stainless steel 316 (SS 316);

(b) reactor coolant, which is sodium; corrosion products are ignored in these calculations as they are presumably included in *(a)*;

(c) actinides that are discharged with the fuel and which pass through the separation processes to be included in the waste;

(d) fission products discharged with the wastes;

(e) the activities of the covergas above the sodium pool (argon in this case).

Before addressing each of the above areas individually, it is worthwhile noting the following assumptions of our reference case:

(1) The fuel is processed 200 days after discharge;

(2) The specific power is 94 MW(th)/t of heavy metal;

(3) The burn-up is 77,500 MW(th)·day/t;

(4) The neutron flux is $4.5 \cdot 10^{15}$ n/cm^2·sec;

(5) The fuel inserted in the reactor consists of:

U235	0.2%
U238	91.7%
Pu238	0.06%
Pu239	4.8%
Pu240	1.8%
Pu241	1%
Pu242	0.12%

(6) The amount of sodium coolant was estimated to be 1.28 m^3/MW(th) for a pool-type LMFBR.

2.1.1 Radioactivity Associated With the Steel Structure

A detailed calculation of the radioactive inventories in LMFBRs and their development over time after reactor shut-down has been performed in [V-1]. A recent recalculation for a 1300 MW(e) plant [V-2] used a more detailed neutron-flux mapping of the various parts of the LMFBR core and primary-vessel system, as shown in Figure V-3 and Table V-I.

In general, the saturation activities are given; however, the activation time used for Co60 and Mn54 is 20 years. The overall activity of the structural material amounts to $2 \cdot 10^8$ Ci per GW(e). The decay-heat production of structural steel after reactor shut-down is 2.5 MW(th), which is about 180 kW(th) per GW(e) after 180 days.

The activities of each of the isotopes in the steel of the primary vessel system are shown in Table V-II. The decay-heat production after 20 years of reactor operation is presented in the same fashion in Table V-III.

Table V-I: Structural Material and Neutron Flux in an LMFBR Primary Vessel System

Category	Component	Steel	Neutron Flux ($n/cm^2 sec$)	Mass (t)
A	Fuel elements	1.4970[a]	$4.5 \cdot 10^{15}$	30
B	Blanket elements	1.4970	$7.5 \cdot 10^{14}$	30
C + D	Reflector elements, parts of fuel elements	1.4970	$5 \cdot 10^{13}$	195
E	Grid plate	1.4948[b]	$3 \cdot 10^{10}$	270
F	Shield tank	1.4948	$5 \cdot 10^{11}$	190
G	Reactor vessel	1.4948	$7.5 \cdot 10^{8}$ to $1 \cdot 10^{7}$	1070

[a] 1.4970 ≙ SS 316 (15% Cr, 1.5% Mn, 15% Ni, 0.05% Co, 1.2% Mo, 0.02% Ta, 0.1% W, balance Fe);

[b] 1.4948 ≙ SS 304 (17.5% Cr, 1.5% Mn, 10.5% Ni, 0.05% Co, 0.9% Mo, 0.02% Ta, 0.1% W, balance Fe).

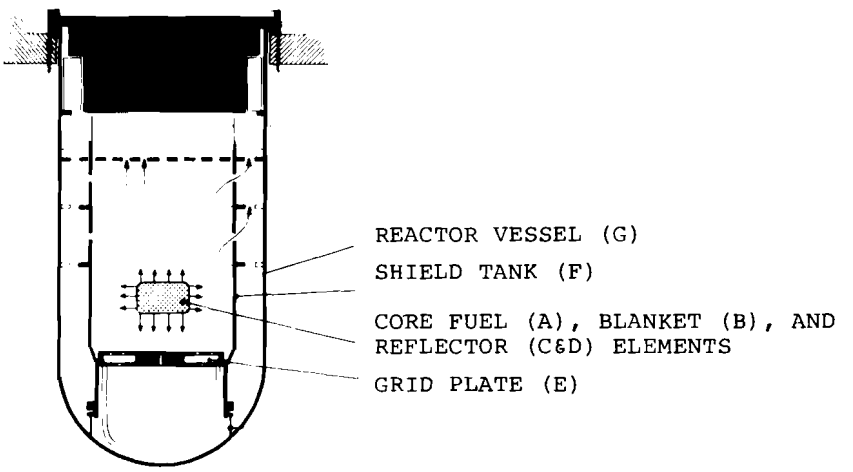

Figure V-3: LMFBR Core and Primary Vessel System

Table V-II: Activities of Structural Material of the Primary Vessel System of a 1300 MW(e) LMFBR (Ci)

Isotope	Zone A	B	C	D	E	F	G	Total	%
Cr51	$1.0 \cdot 10^{7}$	$2.8 \cdot 10^{7}$	$1.7 \cdot 10^{6}$	$2.7 \cdot 10^{6}$	$5.1 \cdot 10^{4}$	$1.7 \cdot 10^{5}$	$1.4 \cdot 10^{3}$	$1.7 \cdot 10^{7}$	6.6
Mn54	$1.2 \cdot 10^{7}$	$1.2 \cdot 10^{7}$	$2.0 \cdot 10^{6}$	$4.1 \cdot 10^{6}$	5.7	1.3	-	$1.3 \cdot 10^{7}$	5.1
Mn56	$5.7 \cdot 10^{7}$	$2.7 \cdot 10^{7}$	$1.6 \cdot 10^{7}$	$2.6 \cdot 10^{7}$	$2.5 \cdot 10^{5}$	$5.4 \cdot 10^{5}$	$2.5 \cdot 10^{3}$	$1.3 \cdot 10^{8}$	50.5
Co57	$1.2 \cdot 10^{5}$		$8.0 \cdot 10^{4}$		$1.5 \cdot 10^{5}$	$2.0 \cdot 10^{5}$		$2.2 \cdot 10^{5}$	0.0
Co58	$5.4 \cdot 10^{7}$	$5.0 \cdot 10^{7}$	$9.3 \cdot 10^{4}$	$1.4 \cdot 10^{5}$	$1.7 \cdot 10^{4}$	4.1	-	$5.9 \cdot 10^{7}$	22.9
Fe59 (from Co59)	$7.8 \cdot 10^{5}$	$3.2 \cdot 10^{6}$	$2.0 \cdot 10^{5}$	$3.2 \cdot 10^{5}$	$3.2 \cdot 10^{4}$	$7.0 \cdot 10^{3}$	$3.0 \cdot 10^{1}$	$1.6 \cdot 10^{6}$	0.6
Co60 (from Ni60)	$2.6 \cdot 10^{5}$	$5.0 \cdot 10^{6}$	$4.7 \cdot 10^{5}$	$3.8 \cdot 10^{6}$	$4.3 \cdot 10^{4}$	$1.0 \cdot 10^{6}$	$3.2 \cdot 10^{4}$	$5.2 \cdot 10^{6}$	2.0
Co60	$1.9 \cdot 10^{5}$	$2.5 \cdot 10^{5}$	$2.8 \cdot 10^{2}$	-	-	-	-	$2.2 \cdot 10^{5}$	0.0
Mo99	$1.0 \cdot 10^{7}$	$4.0 \cdot 10^{6}$	$1.3 \cdot 10^{6}$	$2.3 \cdot 10^{6}$	$1.3 \cdot 10^{4}$	$1.5 \cdot 10^{5}$	$5.4 \cdot 10^{1}$	$1.8 \cdot 10^{7}$	7.0
Ta181	$2.5 \cdot 10^{6}$	$9.7 \cdot 10^{5}$	$1.5 \cdot 10^{6}$	$2.4 \cdot 10^{6}$	$4.1 \cdot 10^{5}$	$1.2 \cdot 10^{6}$	$7.0 \cdot 10^{2}$	$7.5 \cdot 10^{7}$	2.9
W187	$1.0 \cdot 10^{6}$	$3.8 \cdot 10^{5}$	$1.7 \cdot 10^{6}$	$2.7 \cdot 10^{6}$	$4.6 \cdot 10^{4}$	$1.3 \cdot 10^{6}$	$6.2 \cdot 10^{2}$	$6.0 \cdot 10^{6}$	2.3
Total	$1.5 \cdot 10^{8}$	$4.2 \cdot 10^{7}$	$2.3 \cdot 10^{7}$	$4.0 \cdot 10^{7}$	$4.4 \cdot 10^{5}$	$1.2 \cdot 10^{7}$	$5.5 \cdot 10^{3}$	$2.6 \cdot 10^{8}$	100

Table V-III: Decay-heat of Structural Material of the Primary Vessel System of a 1300 MW(e) LMFBR at the End of 20 Years of Operation (kW(th))

Isotope	Zone A	B	C	D	E	F	G	Total	%
Cr51	1.7	0.5	0.3	0.5	-	-	-	3.0	0.1
Mn54	59.4	5.9	0.1	0.2	-	-	-	65.6	2.6
Mn56	817.6	387.3	229.5	373.0	3.6	7.7	-	1818.8	72.3
Co58	361.7	33.5	0.6	0.9	-	-	-	396.8	15.8
Fe59 (from Co59)	6.1	2.5	1.6	2.5	-	0.1	-	12.7	0.5
Co60 (from Ni60)	4.0	7.7	7.2	58.6	0.7	1.5	-	79.7	3.1
Co60	2.9	0.4	-	-	-	-	-	3.3	0.1
Mo99	31.8	12.7	4.1	7.3	-	0.1	-	56.0	2.2
Ta181	19.6	7.6	11.8	20.3	0.3	0.9	-	60.5	2.4
W187	3.9	1.5	6.7	10.6	0.2	0.5	-	23.3	0.9
Total	1308.7	459.6	261.9	473.9	4.8	10.8	-	2519.7	100

The radioactivity in the metallic components stems mainly from the cladding on the fuel elements in the core (approximately 60 per cent), whereas roughly 40 per cent of the activity comes from the blanket and reflector elements. About 0.5 per cent comes form the shield tank, and around 0.2 per cent from the grid plate. It is significant to note that the activity of the reactor vessel is only about $2 \cdot 10^{-5}$ of the total activity.

The density of radioactivity is given in Table V-IV at reactor shut-down; it reveals that, aside from the expected high values for the fuel-element cladding, the density drops off drastically as one moves away from the core. The density of activity of the pressure vessel is actually of the same order as that of natural uranium metal.

Finally, the activity of the steel components is dominated by two isotopes: Mn56 (whose half-life is 2.9 hours), and Co58 (whose half-life is 71.3 days). These two isotopes comprise about 75 per cent of the total shut-down activity.

Table V-IV: Density of Radioactivity of Structural Material of the Primary Vessel System of a 1300 MW(e) LMFBR

Component	Specific Radioactivity (Ci/g)	(Ci/cm^3)
Fuel elements	5	39
Blanket elements	1.4	11
Reflector elements	$3.2 \cdot 10^{-1}$	2.5
Grid plate	$1.6 \cdot 10^{-3}$	$1.3 \cdot 10^{-2}$
Shield tank	$6.3 \cdot 10^{-3}$	$4.9 \cdot 10^{-2}$
Reactor vessel	$5.5 \cdot 10^{-6}$	$4 \cdot 10^{-5}$
Natural Uranium Metal	$3.2 \cdot 10^{-7}$	$6 \cdot 10^{-6}$

2.1.2 Radioactivity Associated With the Fuel

Table V-V gives the activity of the actinides in the fuel elements at the time of discharge and after 30 days of decay. It is important to note that approximately 94 per cent of the total activity in the actinides is due to two radioisotopes: U239 (with a half-life of 23 minutes), and Np239 (with a half-life of 2.4 days).

In practice, discharge of the reactor core will be performed once a year, and at each discharge either one-third or one-half of the core fuel elements will be replaced. In case of a refueling fraction of one-half, for instance, the reactor will on an average contain 50 per cent of fuel that has experienced one-fourth of its target burn-up, and 50 per cent that has experienced three-fourths.

Table V-V: LMFBR Fuel Activity at the Time of Discharge and 30 Days Later[a]

	Half-life	Activity (Ci/t_{HM}) t = 0	t = 30 days
U235	$7.0 \cdot 10^{8}$ yr	$2.0 \cdot 10^{-3}$	$2.0 \cdot 10^{-3}$
U236	$2.3 \cdot 10^{7}$ yr	$1.4 \cdot 10^{-2}$	$1.4 \cdot 10^{-2}$
U237	6.8 day	$3.8 \cdot 10^{5}$	$1.8 \cdot 10^{4}$
U238	$4.5 \cdot 10^{9}$ yr	$2.5 \cdot 10^{-1}$	$2.5 \cdot 10^{-1}$
U239	23 min	$6.6 \cdot 10^{7}$	0
Np239	2.4 day	$6.6 \cdot 10^{7}$	$9.7 \cdot 10^{3}$
Pu238[b]	88 yr	$1.9 \cdot 10^{4}$	$1.9 \cdot 10^{4}$
Pu239	$2.4 \cdot 10^{4}$ yr	$6.1 \cdot 10^{3}$	$6.1 \cdot 10^{3}$
Pu240	$6.5 \cdot 10^{3}$ yr	$9.6 \cdot 10^{3}$	$9.6 \cdot 10^{3}$
Pu241	15 yr	$1.2 \cdot 10^{6}$	$1.2 \cdot 10^{6}$
Pu242	$3.9 \cdot 10^{5}$ yr	$2.9 \cdot 10^{1}$	$2.9 \cdot 10^{1}$
Pu243	5 h	$7.4 \cdot 10^{5}$	0
Am241	$4.3 \cdot 10^{2}$ yr	$4.6 \cdot 10^{3}$	$4.7 \cdot 10^{3}$
Am242m[c]	$1.5 \cdot 10^{2}$ yr	$3.0 \cdot 10^{2}$	$3.0 \cdot 10^{2}$
Am242	16 h	$3.2 \cdot 10^{5}$	$3.0 \cdot 10^{2}$
Am243	$7.4 \cdot 10^{3}$ yr	$1.3 \cdot 10^{2}$	$1.3 \cdot 10^{2}$
Cm242	$1.6 \cdot 10^{2}$ yr	$2.0 \cdot 10^{5}$	$1.8 \cdot 10^{5}$
Cm244	18 yr	$4.0 \cdot 10^{3}$	$4.0 \cdot 10^{3}$
Total Heavy Metal Activity[d]		$1.4 \cdot 10^{8}$	$1.4 \cdot 10^{6}$

[a] Burn-up: 77.5 MW(th)·day/kg; specific power 94 MW(th)/t_{HM}; a 1 GW(e) core has a PuO_2/UO_2 mass content of 30 t, and a heavy-metal content of about 26.7 t (HM refers to heavy metal); irradiation time is two to three years;

[b] Fresh fuel contains 1 per cent Pu238 in plutonium;

[c] Metastable state;

[d] This includes nuclides not shown in the table.

This means, that for all nuclides that experience a reasonably linear build-up, the activity will in average be half of that shown in the Table. But this does not apply to U239 and Np239. U239 will reach its saturation activity within the first day of irradiation, and Np239 within the first one or two months. Therefore, the activities given in Table V-V may also be taken as valid for a reactor of average operation fuel burn-up. That means that the fuel activity for a typical fast-breeder reactor in operation amounts to about $1.4 \cdot 10^8 Ci/t_{HM}$ or $3.7 \cdot 10^9 Ci/GW(e)$ if 30 t of PuO_2/UO_2 fuel or 26.7 t of heavy metal are used.

It is worth noting that the radioactivity of fresh fuel is almost negligible compared to that of the irradiated material.

Because the nuclides that contribute most heavily to the total activity of the fuel have short half-lives, the fuel activity decreases quite rapidly during the first month of the discharge from the reactor; for the case shown, the activity decreases by a factor of 100 within the first 30 days after discharge.

The activities shown in the Table refer to the fuel section of the core, and the activities in the blanket sections are much lower. After the nuclides with very short half-lives have decayed, the blanket activities may total ten per cent of the corresponding value for the core.

2.1.3 Radioactivity Associated With the Coolant

In helium-cooled fast-breeder reactors the coolant activity is negligible. In the case of the LMFBR, contributions from the isotopes Na22 and Na24 must be considered.

The coolant inventory (sodium) in the primary coolant system for a 1000 MW(e) LMFBR varies between 800 m^3 for loop-type and 3200 m^3 for pool-type reactors [V-1].

The total activities in each coolant system are the following:

	Half-life	Activity
Na22	2.6 yr	$4 \cdot 10^3 Ci/GW(e)$
Na24	15 h	$2 \cdot 10^8 Ci/GW(e)$

2.1.4 Radioactivity Associated With Fission Products

The calculation of fission-product inventories is very complicated and the results are difficult to report, because no single isotope accounts for more than two per cent of the total activity. The isotope generation and depletion code ORIGEN [V-3] was used to calculate the activity for each of the blankets and the core region. The results are given in Table V-VI which lists the isotope, its total activity, and half-life. The total activity of the fission products at shut-down is approximately $1 \cdot 10^{10}$ Ci/GW(e) or approximately 4000 Ci/kW(th). This is about a factor of three greater than the activity of the core fuel at discharge.

Volatile fission products like Kr85, I129, I131, and Xe133 will be collected in the fission gas plenum of the fast-reactor fuel elements. The liquid and solid fission products remain within the PuO_2/UO_2 fuel and must be separated during reprocessing.

2.1.5 Radioactivity Associated With the Covergas (Argon)

Argon is used as covergas above the sodium pool in the primary vessel in order to protect the vessel cover against too high temperatures, thermoshocks, etc. In a pool-type LMFBR, up to 400 m^3 of argon is used. Loop-type LMFBRs use less than half. Argon is activated in the neutron field to about 0.02 mCi/cm^3 , yielding a radioactivity of 8 kCi/GW(e) for the covergas, which is negligible compared to that of the coolant.

2.1.6 Decay of Radioactivity of Activated Reactor Components

The decay of the radioactivity associated with all the activated components of our reference LMFBR is shown in Figure V-4. The fission products dominate the radioactivity levels for the first few decades, after which the Pu isotopes are the major factor. Most of the activity of the coolant decays in a few days, while the activity in the steel requires approximately 300 years to decay to the 10^{-7} Ci/W(th) level.

2.2 Mass Flow and Radioactive Inventories in the Reference Breeder Reactor Fuel Cycle

As shown in Figure V-2, the fast-reactor core is only one part of the fuel cycle, and one must properly account for the fact that some of the radioisotopes (i.e. Pu) are recycled back into the reactor for burn-up. We, therefore, must establish the mass flow of radioactive materials through the entire fuel cycle, and then determine the respective radioactive inventories. This is particularly important for the different processes in the reprocessing and long-term waste-storage scheme.

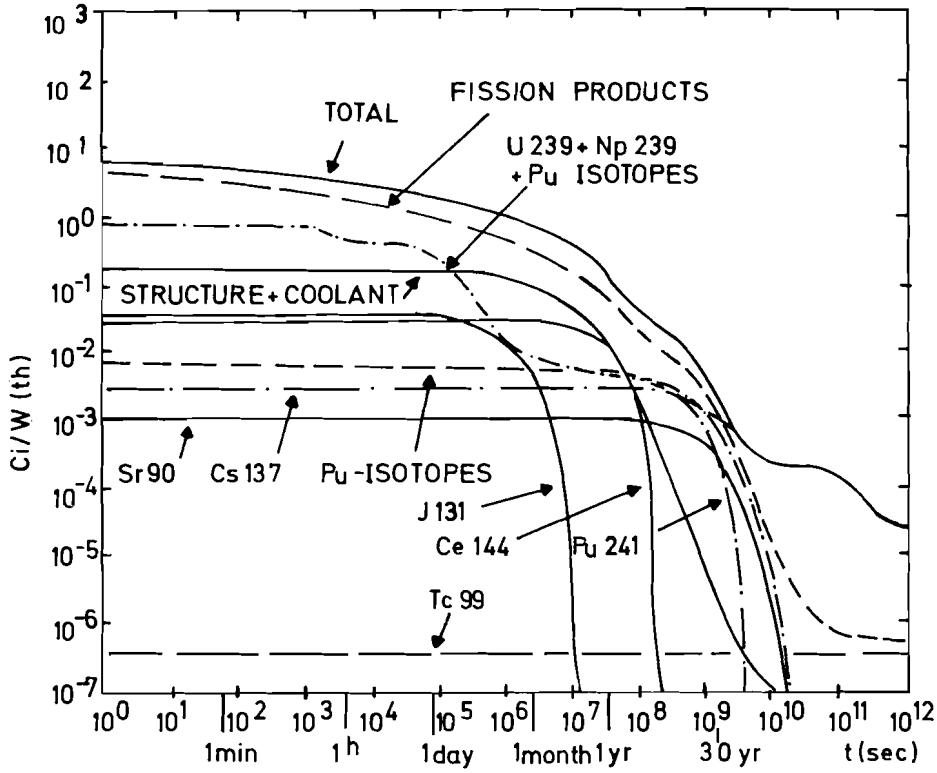

Figure V-4: Specific Radioactivity of Reactor Components as a Function of Time After Discharge from the Fast Reactor Core (without reprocessing)

(a) Core and blanket inventories per GW(e)

In a fast-reactor core, the following material masses are needed, assuming a load factor of one:

Core:

$$3.83 \text{ t } PuO_2 \mathrel{\hat{=}} 3.39 \text{ t}_{HM} \text{ Pu/GW(e) (heavy metal)}$$

$$30 \text{ t } PuO_2/UO_2 - 3.83 \text{ t } PuO_2 = 26.17 \text{ t } UO_2$$

$$\mathrel{\hat{=}} 23.1 \text{ t}_{HM} \text{ U/GW(e) (heavy metal)}$$

Table V-VI: Fission Product Inventory of a Typical 1000 MW(e) LMFBR

Isotope	Activity (Ci)	Half-life	Isotope	Activity (Ci)	Half-life
Zn72	$7.77 \cdot 10^{1}$	1.9 day	Kr85	$3.90 \cdot 10^{5}$	10.76 yr
Ga72	$7.77 \cdot 10^{1}$	14.1 h	Se86	$1.50 \cdot 10^{7}$	16 sec
Zn73	$5.17 \cdot 10^{2}$	3 min	Br86	$1.94 \cdot 10^{7}$	54 sec
Ga73	$5.17 \cdot 10^{2}$	4.8 h	Rb86	$2.89 \cdot 10^{2}$	18.7 day
Ga74	$1.76 \cdot 10^{3}$	7.9 min	Se87	$1.21 \cdot 10^{7}$	10 sec
As74	$9.61 \cdot 10^{-7}$	17.7 day	Br87	$2.34 \cdot 10^{7}$	55.4 sec
Ga75	$6.21 \cdot 10^{3}$	1.9 min	Kr87	$2.38 \cdot 10^{7}$	1.3 min
Ge75m	$2.50 \cdot 10^{2}$	48 sec	Rb87	$7.83 \cdot 10^{-4}$	$4.7 \cdot 10^{10}$ yr
Ge75	$6.21 \cdot 10^{3}$	1.4 min	Br88	$2.51 \cdot 10^{7}$	16 sec
Ga76	$1.87 \cdot 10^{4}$	32 sec	Kr88	$2.89 \cdot 10^{7}$	2.8 h
Ga77	$2.29 \cdot 10^{5}$	17.1 sec	Rb88	$3.05 \cdot 10^{7}$	17.8 min
Ge77m	$2.05 \cdot 10^{5}$	54 sec	As89	$7.18 \cdot 10^{4}$	1 sec
Ge77	$8.08 \cdot 10^{4}$	11.3 h	Se89	$3.56 \cdot 10^{6}$	2 sec
As77	$2.37 \cdot 10^{5}$	1.6 day	Br89	$2.37 \cdot 10^{7}$	4.5 sec
Se77m	$4.03 \cdot 10^{3}$	17.5 sec	Kr89	$3.89 \cdot 10^{7}$	3.1 min
Ga78	$6.10 \cdot 10^{5}$	4 sec	Rb89	$4.00 \cdot 10^{7}$	15 min
Ge78	$6.90 \cdot 10^{5}$	1.5 h	Sr89	$4.00 \cdot 10^{7}$	50.5 day
As78	$6.91 \cdot 10^{5}$	1.5 h	Y 89m	$3.60 \cdot 10^{3}$	16 sec
As79	$1.25 \cdot 10^{6}$	9 min	Br90	$1.51 \cdot 10^{7}$	1.6 sec
Se79m	$1.25 \cdot 10^{6}$	3.9 min	Kr90	$4.27 \cdot 10^{7}$	32 sec
Se79	$2.87 \cdot 10^{1}$	$6.5 \cdot 10^{4}$ yr	Rb90m	$2.91 \cdot 10^{6}$	2.6 min
As80	$2.07 \cdot 10^{6}$	15 sec	Rb90	$4.86 \cdot 10^{7}$	4.3 min
Br80	$1.62 \cdot 10^{5}$	17.6 min	Sr90	$2.52 \cdot 10^{6}$	28.1 yr
As81	$3.42 \cdot 10^{6}$	33 sec	Y 90m	$4.10 \cdot 10^{1}$	3.19 h
Se81m	$6.65 \cdot 10^{3}$	57 min	Y 90	$2.51 \cdot 10^{6}$	2.67 day
Se81	$3.43 \cdot 10^{6}$	18 min	Kr91	$3.43 \cdot 10^{7}$	8.4 sec
Br82	$1.82 \cdot 10^{5}$	1.5 day	Rb91	$5.80 \cdot 10^{7}$	57.4 sec
As83	$6.65 \cdot 10^{6}$	9 sec	Sr91	$9.30 \cdot 10^{7}$	9.7 h
Se83m	$4.65 \cdot 10^{6}$	69 sec	Y 91m	$5.49 \cdot 10^{7}$	50 min
Se83	$2.80 \cdot 10^{6}$	23 min	Y 91	$9.30 \cdot 10^{7}$	58.8 day
Br83	$7.46 \cdot 10^{6}$	2.4 h	Kr92	$2.30 \cdot 10^{7}$	1.0 sec
Kr83m	$7.46 \cdot 10^{6}$	1.9 h	Rb92	$6.16 \cdot 10^{7}$	4.4 sec
Se84	$1.20 \cdot 10^{7}$	3.3 min	Sr92	$7.34 \cdot 10^{7}$	2.71 h
Br84m	$8.08 \cdot 10^{4}$	6 min	Y 92	$7.49 \cdot 10^{7}$	3.5 h
Br84	$1.21 \cdot 10^{7}$	32 min	Kr93	$1.41 \cdot 10^{7}$	1.2 sec
Rb84	$4.71 \cdot 10^{-2}$	33 day	Rb93	$5.31 \cdot 10^{7}$	5.0 sec
As85	$5.59 \cdot 10^{6}$	2.2 sec	Sr93	$8.08 \cdot 10^{7}$	8 min
Se85	$1.30 \cdot 10^{7}$	39 sec	Y 93	$8.40 \cdot 10^{7}$	10.2 h
Br85	$1.41 \cdot 10^{7}$	3 min	Zr93	$8.53 \cdot 10^{1}$	$1.5 \cdot 10^{6}$ yr
Kr85m	$1.41 \cdot 10^{7}$	4.4 h	Nb93m	$4.30 \cdot 10^{0}$	13.6 yr

Table V-VI: Fission Product Inventory of a Typical 1000 MW(e) LMFBR (cont'd)

Isotope	Activity (Ci)	Half-life	Isotope	Activity (Ci)	Half-life
Kr94	$7.66 \cdot 10^{6}$	1 sec	Tc102	$1.30 \cdot 10^{8}$	5.3 sec
Rb94	$3.76 \cdot 10^{7}$	2.7 sec	Rh102m	$4.01 \cdot 10^{-1}$	202 day
Sr94	$8.71 \cdot 10^{7}$	1.8 min	Mo103	$1.21 \cdot 10^{8}$	1.05 min
Y 94	$9.85 \cdot 10^{7}$	20 min	Tc103	$1.25 \cdot 10^{8}$	50 sec
Nb94	$3.89 \cdot 10^{-3}$	$2.0 \cdot 10^{4}$ yr	Ru103	$1.25 \cdot 10^{8}$	39.5 day
Kr95	$5.03 \cdot 10^{6}$	0.8 sec	Rh103m	$1.23 \cdot 10^{8}$	57 min
Rb95	$2.59 \cdot 10^{7}$	0.36 sec	Mo104	$1.07 \cdot 10^{8}$	1.1 min
Sr95	$8.14 \cdot 10^{7}$	26 sec	Tc104	$1.21 \cdot 10^{8}$	18 min
Y 95	$1.13 \cdot 10^{8}$	10.9 min	Rh104	$3.94 \cdot 10^{3}$	42 sec
Zr95	$1.15 \cdot 10^{8}$	65.5 day	Mo105	$7.81 \cdot 10^{7}$	42 sec
Nb95m	$2.30 \cdot 10^{6}$	3.75 day	Tc105	$1.06 \cdot 10^{8}$	7.7 min
Nb95	$1.15 \cdot 10^{8}$	35 day	Ru105	$1.07 \cdot 10^{8}$	4.4 h
Rb96	$2.66 \cdot 10^{6}$	0.23 sec	Rh105m	$2.26 \cdot 10^{7}$	45 sec
Sr96	$4.13 \cdot 10^{7}$	4 sec	Rh105	$1.07 \cdot 10^{8}$	1.48 day
Y 96	$9.79 \cdot 10^{7}$	2.3 min	Tc106	$9.20 \cdot 10^{7}$	37 sec
Nb96	$1.09 \cdot 10^{5}$	23.4 h	Ru106	$7.39 \cdot 10^{7}$	1 yr
Kr97	$7.79 \cdot 10^{2}$	1 sec	Rh106	$7.39 \cdot 10^{7}$	30 sec
Rb97	$3.97 \cdot 10^{5}$	2 sec	Tc107	$6.43 \cdot 10^{7}$	29 sec
Sr97	$1.56 \cdot 10^{7}$	3 sec	Ru107	$7.41 \cdot 10^{7}$	4.2 min
Y 97	$7.84 \cdot 10^{7}$	6 sec	Rh107	$7.41 \cdot 10^{7}$	22 min
Zr97	$1.08 \cdot 10^{8}$	16.8 h	Pd107	$1.59 \cdot 10^{1}$	$7.0 \cdot 10^{6}$ yr
Mb97m	$1.04 \cdot 10^{8}$	1 min	Tc108	$3.82 \cdot 10^{7}$	10 sec
Nb97	$1.10 \cdot 10^{8}$	1.23 h	Ru108	$5.40 \cdot 10^{7}$	4.5 min
Zr98	$1.16 \cdot 10^{8}$	30.7 sec	Rh108	$5.40 \cdot 10^{7}$	17 sec
Nb98m	$1.16 \cdot 10^{8}$	2.9 sec	Tc109	$1.22 \cdot 10^{7}$	10 sec
Nb98	$5.12 \cdot 10^{6}$	51 min	Ru109	$3.21 \cdot 10^{7}$	34.5 sec
Tc98	$6.68 \cdot 10^{-5}$	$1.5 \cdot 10^{6}$ yr	Rh109	$3.54 \cdot 10^{7}$	30 sec
Y 99	$2.47 \cdot 10^{7}$	0.8 sec	Pd109	$3.42 \cdot 10^{7}$	13.5 h
Zr99	$9.62 \cdot 10^{7}$	1.6 sec	Ag109m	$3.42 \cdot 10^{7}$	40 sec
Nb99m	$4.27 \cdot 10^{7}$	10 sec	Rh110	$1.88 \cdot 10^{7}$	5 sec
Nb99	$8.12 \cdot 10^{7}$	2.4 min	Rh111	$7.30 \cdot 10^{6}$	1 sec
Mo99	$1.28 \cdot 10^{8}$	2.78 day	Pd111m	$7.44 \cdot 10^{4}$	5.5 h
Tc99m	$1.13 \cdot 10^{8}$	6 h	Pd111	$7.44 \cdot 10^{6}$	22 min
Tc99	$9.07 \cdot 10^{2}$	$2.1 \cdot 10^{5}$ yr	Ag111m	$7.46 \cdot 10^{6}$	1.2 min
Nb100	$1.27 \cdot 10^{8}$	2.8 min	Ag111	$7.46 \cdot 10^{6}$	7.5 day
Tc100	$9.73 \cdot 10^{4}$	17 sec	Pd112	$4.36 \cdot 10^{6}$	21 h
Nb101	$1.18 \cdot 10^{8}$	1 min	Ag112	$4.36 \cdot 10^{6}$	3.2 h
Mo101	$1.31 \cdot 10^{8}$	14.6 min	Pd113	$1.96 \cdot 10^{6}$	1.5 min
Tc101	$1.31 \cdot 10^{8}$	14 min	Ag113m	$1.96 \cdot 10^{6}$	1.2 min
Mo102	$1.29 \cdot 10^{8}$	11.5 min	Ag113	$9.80 \cdot 10^{5}$	5.3 h

Table V-VI: Fission Product Inventory of a Typical 1000 MW(e) LMFBR (cont'd)

Isotope	Activity (Ci)	Half-life		Isotope	Activity (Ci)	Half-life	
Cd113m	$6.89 \cdot 10^{0}$	13.6	yr	Sn125	$2.90 \cdot 10^{6}$	9.6	day
Pd114	$2.14 \cdot 10^{6}$	2.4	min	Sb125	$1.32 \cdot 10^{6}$	2.7	yr
Ag114	$2.34 \cdot 10^{6}$	5.2	sec	Te125m	$2.63 \cdot 10^{5}$	58	day
Pd115	$1.75 \cdot 10^{6}$	38	sec	Sn126	$7.24 \cdot 10^{1}$	$1.0 \cdot 10^{5}$	yr
Ag115m	$4.89 \cdot 10^{5}$	49	sec	Sb126m	$7.24 \cdot 10^{1}$	19	min
Ag115	$1.88 \cdot 10^{6}$	20	min	Sb126	$1.82 \cdot 10^{5}$	2.4	day
Cd115m	$2.54 \cdot 10^{5}$	43	day	J 126	$2.14 \cdot 10^{0}$	12.8	day
Cd115	$1.96 \cdot 10^{6}$	2.23	day	Sn127	$6.91 \cdot 10^{6}$	2.1	h
In115m	$1.96 \cdot 10^{6}$	4.5	h	Sb127	$7.96 \cdot 10^{6}$	3.8	day
In115	$4.62 \cdot 10^{-9}$	$6 \cdot 10^{14}$	yr	Te127m	$1.30 \cdot 10^{6}$	109	day
Pd116	$1.34 \cdot 10^{6}$	30	sec	Te127	$7.90 \cdot 10^{6}$	9.4	h
Ag116	$1.64 \cdot 10^{6}$	2.5	min	Sn128	$9.73 \cdot 10^{6}$	59	min
Pd117	$9.83 \cdot 10^{5}$	5	sec	Sb128m	$9.50 \cdot 10^{6}$	10	min
Ag117	$1.55 \cdot 10^{6}$	42	sec	Sb128	$4.07 \cdot 10^{6}$	9.3	h
Cd117m	$7.75 \cdot 10^{5}$	3.1	h	J 128	$1.95 \cdot 10^{2}$	25	min
Cd117	$1.29 \cdot 10^{6}$	2.5	h	In129	$1.65 \cdot 10^{6}$	8.8	min
In117m	$1.60 \cdot 10^{6}$	1.95	h	Sn129m	$4.94 \cdot 10^{6}$	2	min
In117	$7.51 \cdot 10^{5}$	38	min	Sn129	$4.95 \cdot 10^{6}$	8.8	min
Ag118	$1.45 \cdot 10^{6}$	5.3	sec	Sb129	$1.53 \cdot 10^{7}$	4.3	h
Cd118	$1.58 \cdot 10^{6}$	49	min	Te129m	$6.54 \cdot 10^{6}$	34.1	day
In118	$1.58 \cdot 10^{6}$	5	sec	Te129	$1.72 \cdot 10^{7}$	1.15	h
Cd119	$1.57 \cdot 10^{6}$	3.4	min	J 129	$1.69 \cdot 10^{0}$	$1.7 \cdot 10^{7}$	yr
In119m	$1.57 \cdot 10^{6}$	18	min	Xe129m	$1.25 \cdot 10^{-1}$	8	day
In119	$9.40 \cdot 10^{4}$	2.3	min	Sn130	$1.55 \cdot 10^{7}$	2.6	min
Sn119m	$1.33 \cdot 10^{6}$	245	day	Sb130m	$7.76 \cdot 10^{6}$	6	min
Cd120	$1.52 \cdot 10^{6}$	1	min	Sb130	$3.40 \cdot 10^{7}$	37	min
In120	$1.57 \cdot 10^{6}$	3.2	sec	J 130	$1.44 \cdot 10^{5}$	12.3	h
Cd121	$1.43 \cdot 10^{6}$	12.8	sec	Cs130	0.0	29.1	min
In121m	$7.14 \cdot 10^{5}$	3.1	min	In131	$8.99 \cdot 10^{5}$	1	sec
In121	$8.98 \cdot 10^{5}$	30	sec	Sn131	$2.11 \cdot 10^{7}$	1.3	min
Sn121m	$4.04 \cdot 10^{2}$	76	yr	Sb131	$7.00 \cdot 10^{7}$	23	min
Sn121	$1.62 \cdot 10^{6}$	1.125	day	Te131m	$1.14 \cdot 10^{7}$	1.25	day
In122	$1.19 \cdot 10^{6}$	8	sec	Te131	$7.37 \cdot 10^{7}$	25	min
Sb122	$6.33 \cdot 10^{4}$	2.68	day	J 131	$8.54 \cdot 10^{7}$	8.05	day
In123	$1.80 \cdot 10^{6}$	6	sec	Xe131m	$5.12 \cdot 10^{5}$	11.8	day
Sn123	$1.93 \cdot 10^{6}$	129	day	Cs131	$1.31 \cdot 10^{-2}$	9.7	day
In124	$1.42 \cdot 10^{6}$	4	sec	Sn132	$1.21 \cdot 10^{7}$	1	min
Sb124	$7.07 \cdot 10^{2}$	60.3	day	Sb132	$4.78 \cdot 10^{7}$	3.13	min
In125	$8.55 \cdot 10^{5}$	17.8	sec	Te132	$8.14 \cdot 10^{7}$	3.25	day
Sn125m	$7.43 \cdot 10^{5}$	9.7	min	J 132	$8.64 \cdot 10^{7}$	2.4	h

Table V-VI: Fission Product Inventory of a Typical 1000 MW(e) LMFBR (cont'd)

Isotope	Activity (Ci)	Half-life	Isotope	Activity (Ci)	Half-life
Cs132	$6.80 \cdot 10^{-1}$	6.5 day	Ba139	$1.19 \cdot 10^{8}$	1.38 h
Sn133	$7.98 \cdot 10^{6}$	55 sec	J 140	$1.05 \cdot 10^{7}$	0.8 sec
Sb133	$4.44 \cdot 10^{7}$	2.7 min	Xe140	$4.49 \cdot 10^{7}$	13.5 sec
Te133m	$3.28 \cdot 10^{7}$	54 min	Cs140	$1.02 \cdot 10^{8}$	1.05 min
Te133	$7.89 \cdot 10^{7}$	12.5 min	Ba140	$1.15 \cdot 10^{8}$	12.8 day
J 133	$1.26 \cdot 10^{8}$	20.8 h	La140	$1.15 \cdot 10^{8}$	1.675 day
Xe133m	$3.02 \cdot 10^{6}$	2.2 day	Xe141	$1.91 \cdot 10^{7}$	1.7 sec
Xe133	$1.26 \cdot 10^{8}$	5.65 day	Cs141	$6.96 \cdot 10^{7}$	24 sec
Sb134	$2.77 \cdot 10^{7}$	1.5 sec	Ba141	$9.57 \cdot 10^{7}$	18 min
Te134	$1.01 \cdot 10^{8}$	43 min	La141	$9.75 \cdot 10^{7}$	3.9 h
J 134	$1.41 \cdot 10^{8}$	52 min	Ce141	$9.75 \cdot 10^{7}$	32.5 day
Cs134	$1.17 \cdot 10^{3}$	2.05 yr	Xe142	$7.84 \cdot 10^{6}$	1.2 sec
Sb135	$1.01 \cdot 10^{7}$	1.6 sec	Cs142	$5.23 \cdot 10^{7}$	2.3 sec
Te135	$7.39 \cdot 10^{7}$	29 sec	Ba142	$1.06 \cdot 10^{8}$	11 min
J 135	$1.36 \cdot 10^{8}$	6.7 h	La142	$1.14 \cdot 10^{8}$	1.54 h
Xe135m	$3.53 \cdot 10^{7}$	15.6 min	Pr142	$7.67 \cdot 10^{1}$	19.2 h
Xe135	$1.55 \cdot 10^{8}$	9.15 h	Xe143	$2.37 \cdot 10^{6}$	1 sec
Cs135m	$1.93 \cdot 10^{4}$	53 min	Cs143	$2.20 \cdot 10^{7}$	1.6 sec
Cs135	$1.18 \cdot 10^{2}$	$2.0 \cdot 10^{6}$ yr	Ba143	$7.42 \cdot 10^{7}$	13.2 sec
La135	$1.91 \cdot 10^{-4}$	19.4 h	La143	$9.48 \cdot 10^{7}$	14 min
Te136	$3.07 \cdot 10^{7}$	33 sec	Ce143	$1.03 \cdot 10^{8}$	1.30 day
J 136	$6.58 \cdot 10^{7}$	1.38 min	Pr143	$1.03 \cdot 10^{8}$	13.6 day
Cs136	$2.63 \cdot 10^{6}$	12.9 day	Xe144	$3.28 \cdot 10^{5}$	1 sec
Sn137	$3.25 \cdot 10^{3}$	1 sec	Cs144	$7.27 \cdot 10^{6}$	1.1 sec
Sb137	$7.81 \cdot 10^{5}$	2 sec	Ba144	$3.73 \cdot 10^{7}$	11.0 sec
Te137	$2.18 \cdot 10^{7}$	3 sec	La144	$7.61 \cdot 10^{7}$	41 sec
J 137	$9.07 \cdot 10^{7}$	24 sec	Ce144	$7.04 \cdot 10^{7}$	284 day
Xe137	$1.31 \cdot 10^{8}$	3.9 min	Pr144	$7.04 \cdot 10^{7}$	17.3 min
Cs137	$7.00 \cdot 10^{6}$	30 yr	Nd144	$3.46 \cdot 10^{-8}$	$2.1 \cdot 10^{15}$ yr
Ba137m	$9.48 \cdot 10^{6}$	2.55 min	Pm144	$9.53 \cdot 10^{-4}$	1 yr
La137	$3.42 \cdot 10^{-5}$	$6.0 \cdot 10^{4}$ yr	Ce145	$6.74 \cdot 10^{7}$	3 min
Ce137m	0.0	1.43 day	Pr145	$6.77 \cdot 10^{7}$	5.98 h
Ce137	0.0	9 h	Ce146	$5.55 \cdot 10^{7}$	13.9 min
J 138	$4.18 \cdot 10^{7}$	6 sec	Pr146	$5.70 \cdot 10^{7}$	24 min
Xe138	$1.13 \cdot 10^{8}$	14.1 min	Pm146	$3.46 \cdot 10^{0}$	5.53 yr
Cs138	$1.33 \cdot 10^{8}$	32.3 min	Ce147	$4.16 \cdot 10^{7}$	1.2 min
La138	$4.75 \cdot 10^{-10}$	$1.1 \cdot 10^{11}$ yr	Pr147	$4.66 \cdot 10^{7}$	12 min
J 139	$1.74 \cdot 10^{7}$	2.7 sec	Nd147	$4.66 \cdot 10^{7}$	11.1 day
Xe139	$7.68 \cdot 10^{7}$	41 sec	Pm147	$2.00 \cdot 10^{7}$	2.62 yr
Cs139	$1.15 \cdot 10^{8}$	9 min	Sm147	$1.48 \cdot 10^{-4}$	$1.08 \cdot 10^{11}$ yr

Table V-VI: Fission Product Inventory of a Typical 1000 MW(e) LMFBR (cont'd)

Isotope	Activity (Ci)	Half-life	
Ce148	$2.72 \cdot 10^{7}$	43	sec
Pr148	$3.69 \cdot 10^{7}$	1.98	min
Pm148	$5.44 \cdot 10^{3}$	5.4	day
Pr149	$2.59 \cdot 10^{7}$	2.3	min
Nd149	$3.02 \cdot 10^{7}$	1.73	h
Pm149	$3.03 \cdot 10^{7}$	2.2	day
Pm150	$1.28 \cdot 10^{5}$	2.7	h
Eu150	0.0	6.2	yr
La151	$2.89 \cdot 10^{4}$	1	sec
Ce151	$1.04 \cdot 10^{6}$	2	sec
Pr151	$6.94 \cdot 10^{6}$	3	sec
Nd151	$1.61 \cdot 10^{7}$	12	min
Pm151	$1.79 \cdot 10^{7}$	1.2	day
Sm151	$3.05 \cdot 10^{5}$	87	yr
Gd151	$8.00 \cdot 10^{-4}$	120	day
Pm152	$1.48 \cdot 10^{7}$	6	min
Eu152	$3.63 \cdot 10^{1}$	12.4	yr
Pm153	$9.27 \cdot 10^{6}$	5.5	min
Sm153	$1.01 \cdot 10^{7}$	1.95	day
Pm154	$6.36 \cdot 10^{6}$	1.6	min
Eu154	$9.60 \cdot 10^{3}$	16	yr
Sm155	$5.10 \cdot 10^{6}$	22.4	min
Eu155	$2.87 \cdot 10^{6}$	1.81	yr
Sm156	$3.24 \cdot 10^{6}$	9.4	h
Eu156	$3.56 \cdot 10^{6}$	15.1	day
Sm157	$1.76 \cdot 10^{6}$	30	sec
Eu157	$2.36 \cdot 10^{6}$	15.1	h
Eu158	$1.35 \cdot 10^{6}$	46	min
Tb158	$9.70 \cdot 10^{1}$	150	yr
Eu159	$7.37 \cdot 10^{5}$	18	min
Gd159	$8.90 \cdot 10^{5}$	18.56	h
Eu160	$3.02 \cdot 10^{5}$	40	sec
Tb160	$1.59 \cdot 10^{4}$	72.1	day
Gd161	$2.30 \cdot 10^{5}$	3.6	min
Tb161	$2.61 \cdot 10^{5}$	6.9	day
Gd162	$1.57 \cdot 10^{3}$	10.4	min
Tb162	$2.55 \cdot 10^{3}$	7.8	min
Tb163	$2.62 \cdot 10^{3}$	19.5	min
Tb164	$6.71 \cdot 10^{2}$	3	min
Dy165	$2.65 \cdot 10^{2}$	2.35	h
Dy166	$1.50 \cdot 10^{2}$	3.4	day
Ho166	$1.70 \cdot 10^{2}$	1.11	day

Axial blankets:

$$18 \text{ t } UO_2 \mathrel{\hat{=}} 15.93 \text{ t}_{HM} \text{ U/GW(e)}$$

Radial blanket:

$$43.1 \text{ t } UO_2 \mathrel{\hat{=}} 38.23 \text{ t}_{HM} \text{ U/GW(e)}$$

(b) Masses of fuel charged to the core and blanket per year and GW(e):

Residency time:

2.2 years for core and axial blankets;
7 years for the radial blanket;

Core:

1.54 t Pu/GW(e)·yr
10.53 t U/GW(e)·yr

} 19.31 t (U+Pu)/GW(e)·yr in core and axial blanket;

Axial blanket:

7.24 t U/GW(e)·yr

Radial blanket:

5.46 t U/GW(e)·yr

(c) Masses discharged per year and GW(e):

Assuming that fissioning of one gram of Pu produces 0.99 MW(th)·day, the total mass of fission products for the generation of one GW(th)·day amounts to 1.01 kg. With a thermal efficiency of 40 per cent, the following masses of fission products are produced per GW(e)·yr:

$$1.01\left(\frac{\text{kg}}{\text{GW(th)}\cdot\text{day}}\right) \cdot 365\left(\frac{\text{day}}{\text{yr}}\right) \cdot 2.5\left(\frac{\text{GW(th)}}{\text{GW(e)}}\right) = 922\left(\frac{\text{kg}}{\text{GW(e)}\cdot\text{yr}}\right)$$

90 per cent of the power is generated in the core and axial blankets, and ten per cent in the radial blanket. This results in 830 kg of fission products that are built up in the core and axial blankets, and 92 kg in the radial blanket, per GW(e) and year.

The heavy metal discharged per year is:

Core and axial blanket:

$$19.31 - 0.830 = 18.48\ t_{HM}/GW(e)\cdot yr$$

Radial blanket:

$$5.46 - 0.092 = 5.37\ t_{HM}/GW(e)\cdot yr$$

$$\Big\} 23.85\ t_{HM}/GW(e)\cdot yr$$

As was shown in [V-4], approximately 1.4 kg (Np+Am+Cm)/t_{HM} is present in spent fuel. So the mass of Np, Am, and Cm in the reactor fuel per GW(e)·yr amounts to:

$$23.85\ \frac{t_{HM}}{GW(e)\cdot yr} \cdot 1.4\ \frac{kg(Np+Am+Cm)}{t_{HM}} = 33\ kg(Np+Am+Cm)/GW(e)\cdot yr$$

More than 90 per cent of these actinides are present in the core. Therefore, the masses for U and Pu discharged per year are:

Core and axial blanket:

$$18.48 - 0.030 = 18.45\ t(U+Pu)/GW(e)\cdot yr$$

Radial blanket:

$$5.37 - 0.003 = 5.37\ t(U+Pu)/GW(e)\cdot yr$$

Core and blankets:

$$23.82\ t(U+Pu)/GW(e)\cdot yr$$

The Pu gain by breeding amounts to about 100 kg/GW(e)·yr* since:

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* This low value is assumed for present prototype oxide breeder reactors. Values up to 150 to 300 kg Pu/GW(e)·yr are anticipated for more advanced FBRs.

$$Pu_{gain} = Pu(out) - Pu(in)$$

$$= 1.64 - 1.54 = 0.1 \text{ t/GW(e)·yr}$$

$$U(out) = 23.82 - 1.64 = 22.18 \text{ t/GW(e)·yr}$$

The masses of heavy metal and fission products charged and and discharged per year are demonstrated in Table V-VII.

The fuel cycle of this 1000 MW(e) LMFBR is shown in Figure V-5 under the assumption that Pu- and U-losses of one per cent each in the reprocessing facility and the fuel fabrication plant, respectively, can be expected.

Table V-VII: Flow of Material per Year in a 1000 MW(e) LMFBR (t/GW(e)·yr)

Material Charged			Material Discharged	
By Region		Total	By Region	Total
Core	Pu: 1.54 U : 10.53	Pu: 1.54 U : 23.23	Core and axial blankets U+Pu : 18.45 Np+Am+Cm : 0.03 fission products: 0.830	Pu : 1.64 U : 22.18 Np+Am+Cm: 0.033 Fission products : 0.922
Axial blanket	U : 7.24			
Radial blanket	U : 5.46		Radial blanket U+Pu : 5.37 Np+Am+Cm : 0.003 fission products: 0.092	

(d) Radioactive inventories

The isotopic composition of spent fuel has been determined recently [V-4] with the isotope generation and depletion code ORIGEN [V-3] for each of the blankets and the core region. The averaged isotopic composition for the discharged fuel was then determined by mixing the fuel from the core and the axial and radial blankets after discharge from the reactor (see Table V-VI).

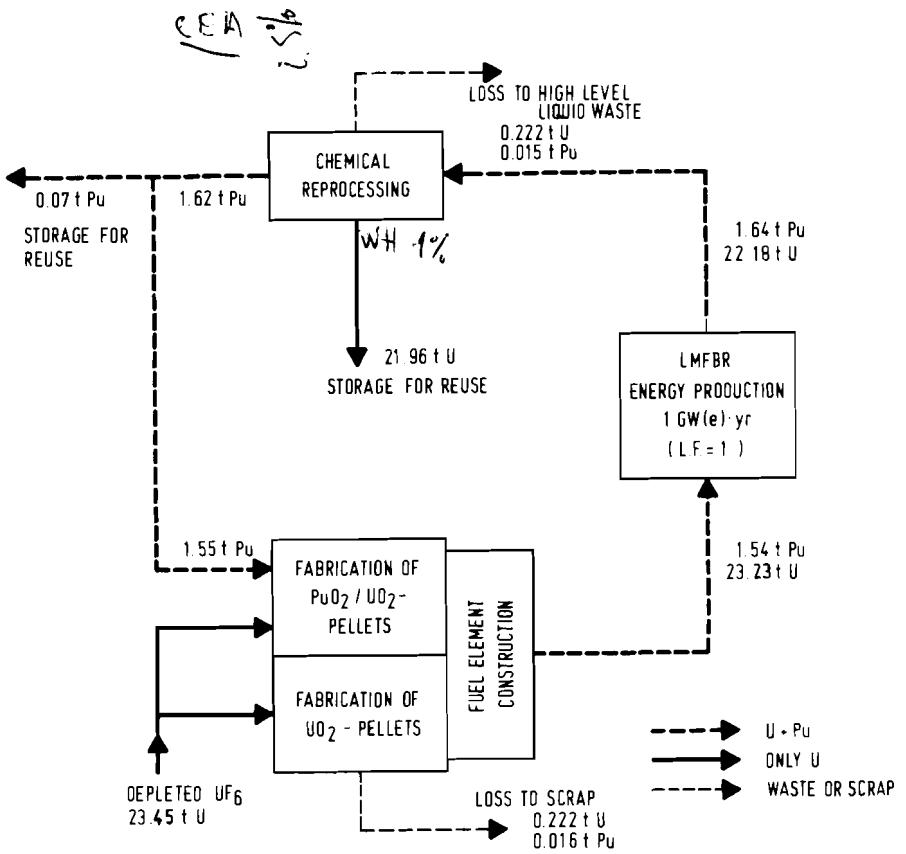

Figure V-5: Integrated LMFBR Fuel Cycle System--Yearly Mass Flows of Uranium and Plutonium for One GW(e)

Table V-VIII shows the summarized results in Ci per ton of heavy metal for various cooling times. The table contains the sum of fission-product and actinide activities as well as some radiologically important fission products.

The high burn-ups for fast-reactor fuels require relatively long cooling times before chemical reprocessing can start. For LWR fuel, 220 days of storage time in water pools at the site of the reactor or the reprocessing plant are foreseen today. Some authors emphasize even longer cooling times [V-5]. For economic reasons, however, the cooling periods for FBR fuel will have to be even shorter. About 200 days of cooling between discharge and reprocessing are proposed at present.

Table V-VIII: Activity of Discharged Fuel (core and axial and radial blankets are mixed) (Ci/t_{HM})

	Isotopes	Cooling Time (days) 150	270	365
Fission Products	Sr90 (β)	$5.22 \cdot 10^{4}$	$5.18 \cdot 10^{4}$	$5.15 \cdot 10^{4}$
	Cs137 (γ)	$1.40 \cdot 10^{5}$	$1.39 \cdot 10^{5}$	$1.38 \cdot 10^{5}$
	I129	$4.35 \cdot 10^{-2}$	$4.35 \cdot 10^{-2}$	$4.35 \cdot 10^{-2}$
	Total Fiss. Prod.	$7.02 \cdot 10^{6}$	$4.45 \cdot 10^{6}$	$3.50 \cdot 10^{6}$
Actinides	Pu (α)	$1.84 \cdot 10^{4}$	$1.85 \cdot 10^{4}$	$1.85 \cdot 10^{4}$
	Am (α)	$2.74 \cdot 10^{3}$	$3.04 \cdot 10^{3}$	$3.28 \cdot 10^{3}$
	Cm (α)	$5.60 \cdot 10^{4}$	$3.43 \cdot 10^{4}$	$2.36 \cdot 10^{4}$
	Total α	$7.72 \cdot 10^{4}$	$5.60 \cdot 10^{4}$	$4.85 \cdot 10^{4}$
	Np239 (β)	$6.35 \cdot 10^{1}$	$6.35 \cdot 10^{1}$	$6.35 \cdot 10^{1}$
	Pu241 (β)	$5.75 \cdot 10^{5}$	$5.65 \cdot 10^{5}$	$5.60 \cdot 10^{5}$
	Total β	$5.80 \cdot 10^{5}$	$5.65 \cdot 10^{5}$	$5.60 \cdot 10^{5}$
	Total Actinides	$6.57 \cdot 10^{5}$	$6.21 \cdot 10^{5}$	$6.09 \cdot 10^{5}$
Total		$7.68 \cdot 10^{6}$	$5.07 \cdot 10^{6}$	$4.11 \cdot 10^{6}$

As a consequence of the chemical separation of the fission products and the actinides Np, Am, and Cm from the uranium and plutonium fuel, the radioactive inventory of the fuel is reduced by one order of magnitude. This is shown in Table V-IX, where the activities are normalized to 1 GW(e)·yr, which means that the specific activities (Ci/t) are multiplied by the masses of Figure V-5 under the assumption of a load factor of one. In the Table the activities are listed for different steps in the fuel cycle: transport from the reactor to the reprocessing plant (150 days after discharge [V-6]), the reprocessing before and after chemical separation (200 days), and fuel fabrication (365 days). The build-up of Am241 through

Table V-IX: Radioactive Composition of the Fuel in the Fuel Cycle
(Ci/GW(e)·yr)

Isotopes	Days After Discharge: 150 (Transportation)	200 (Fuel Reprocessing) (before)	200 (Fuel Reprocessing) (after) (separation)	365 (fabrication)
Fission Products	$1.67 \cdot 10^{8}$	$1.22 \cdot 10^{8}$	-	-
Actinides				
Pu (α)	$4.38 \cdot 10^{5}$	$4.38 \cdot 10^{5}$	$4.34 \cdot 10^{5}$	$4.32 \cdot 10^{5}$
Am (α)	$6.53 \cdot 10^{4}$	$6.84 \cdot 10^{4}$	-	-
Cm (α)	$1.33 \cdot 10^{6}$	$9.53 \cdot 10^{4}$	-	-
Σ α	$1.84 \cdot 10^{6}$	$1.46 \cdot 10^{6}$	$4.34 \cdot 10^{5}$	$4.32 \cdot 10^{5}$
Pu (β)	$1.37 \cdot 10^{7}$	$1.36 \cdot 10^{7}$	$1.34 \cdot 10^{7}$	$1.31 \cdot 10^{7}$
Total	$1.56 \cdot 10^{7}$	$1.52 \cdot 10^{7}$	$1.38 \cdot 10^{7}$	$1.35 \cdot 10^{7}$
Total	$1.83 \cdot 10^{8}$	$1.37 \cdot 10^{8}$	$1.38 \cdot 10^{7}$	$1.35 \cdot 10^{7}$

decay of Pu241 between reprocessing and refabrication can be neglected, because the americium must be separated for reasons of γ-radiation protection [V-7]. After reprocessing, approximately 97 per cent of the residual activity is Pu241 activity (β-activity).

(e) Decay of Radioactive Inventories in Normal Fission Fuel Cycle

We are now in a position to incorporate the effects of fuel reprocessing and reinsertion of Pu isotopes into the fission reactors. Figures V-6 and V-7 show how the long-lived activity decays with time per W(th) both for the fuel and for the high-level waste. There is a major difference between Figure V-4 and Figures V-6 and V-7: It is assumed for the latter that after 200 days, 99 per cent of the Pu isotopes are removed from the fuel waste and recycled into the reactor.

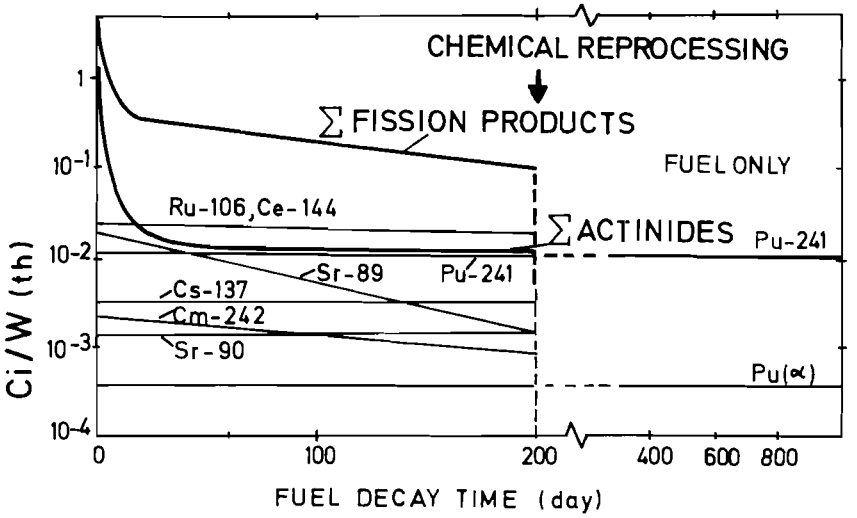

Figure V-6: Specific Radioactivity of LMFBR Fuel as a Function of Time

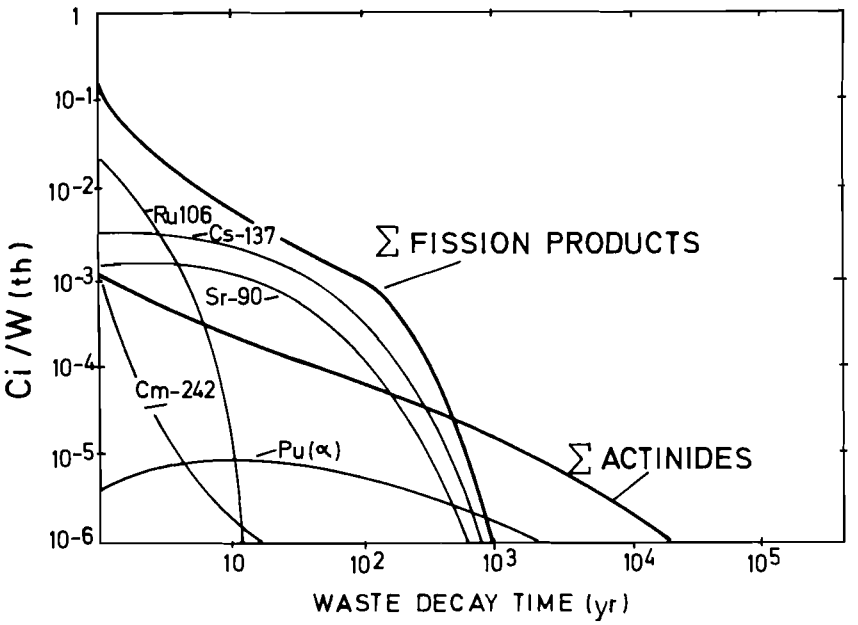

Figure V-7: Specific Radioactivity of LMFBR High Level Waste as a Function of Time

At shut-down, the activity of the fuel is dominated by the fission products, and approximately 25 per cent of the activity comes from the actinides. The activity in the actinides decays away rapidly, so that 30 days after shut-down the fission products produce more than 90 per cent of the activity. At the time of reprocessing the total actinide activity has dropped by about two orders of magnitude, from approximately one Ci/W(th) to approximately 10^{-2} Ci/W(th). Most of the latter is due to Pu241. It is presumed that after a few years the fuel is reinserted into the reactor.

The acivity of the high-level liquid waste ranges from approximately 10^{-1} Ci/W(th) after one year of decay to approximately 10^{-3} Ci/W(th) after 100 years. Roughly 500 years after decay, the actinides that pass through reprocessing are the dominant source of activity for over 10,000 years.

2.3 Biological Hazard Potential Values for Fission Reactors

The radioactivity expressed in curies is an insufficient indicator of the hazard to the public in the case of radioactive release. Comparisons between different nuclear systems can better be accomplished by the use of the biological hazard potential (BHP) unit.

The BHP for the LMFBR was calculated by careful weighing of the results for the core and blanket zones [V-4]. Chemical reprocessing was assumed to take place 200 days after discharge from the reactor. The calculation results for the fuel and waste are given in Tables V-X and V-XI, and displayed in Figures V-8 to V-10. Since approximately all fission products and all actinides (including one per cent of U/Pu) are separated from the discharged fuel, the BHP index of the separated fuel after reprocessing is represented by the BHP of the plutonium alone (plus a negligible contribution from uranium).

It can be seen that, for the inhalation hazard, the actinides dominate even before the fission products are chemically separated (see Figure V-8). The decrease of the inhalation hazard of the actinides, which is due to the separation of Np, Am, Cm, and one per cent U/Pu, is relatively small, since the inhalation hazard is largely determined by that of Pu. The BHP for inhalation if all the radioisotopes were released is roughly 4000 km^3 of air per kW(th)·yr of energy for the first three years after reactor shut-down.

The situation is reversed with respect to ingestion of water. Figure V-8 shows that the fission-product BHP (water) is one to two orders of magnitude higher than that for the actinides (including Pu isotopes). The fission product values range from roughly 0.1 km^3 of water per kW(th)·yr to $4 \cdot 10^{-3}$ km^3 per kW(th)·yr at the time of reprocessing. After that point, the ingestion factors for the fuel remain approximately constant at $6 \cdot 10^{-5}$ km^3 of water per kW(th)·yr up to the time of reinsertion into the reactor.

Table V-X: Biological Hazard Potential of Breeder Reactor Fuel (km^3 of air or water per kW(th)·yr)

	Time After Shut-down (day)					
	0	150	200 before separation	200 after separation	365	1096
Inhalation (air)						
Fission Products	530	133	122	-	-	-
Actinides	5125	5000	5000	4590	4550	4390
Ingestion (water)						
Fission Products	0.092	$4.5 \cdot 10^{-3}$	$4.5 \cdot 10^{-3}$	-	-	-
Actinides	$4.1 \cdot 10^{-3}$	$9.7 \cdot 10^{-5}$	$9.2 \cdot 10^{-5}$	$6.2 \cdot 10^{-5}$	$6.2 \cdot 10^{-5}$	$5.9 \cdot 10^{-5}$

Table V-XI: Biological Hazard Potential of High Level Liquid Waste of Breeder Reactors (km^3 of air or water per kW(th)·yr)

	Time After Shut-down (yr)					
	0	10	100	1000	10,000	100,000
Inhalation (air)						
Fission Products	128	15.6	1.7	$1.3 \cdot 10^{-4}$	$1.2 \cdot 10^{-4}$	$1 \cdot 10^{-4}$
Actinides	396	278	185	52	14	1
Ingestion (water)						
Fission Products	$4.2 \cdot 10^{-3}$	$1.4 \cdot 10^{-3}$	$1.5 \cdot 10^{-4}$	$8 \cdot 10^{-9}$	$8 \cdot 10^{-9}$	$7.7 \cdot 10^{-9}$
Actinides	$3.5 \cdot 10^{-5}$	$1 \cdot 10^{-5}$	$7.3 \cdot 10^{-6}$	$1.8 \cdot 10^{-6}$	$2.2 \cdot 10^{-7}$	$7.2 \cdot 10^{-8}$

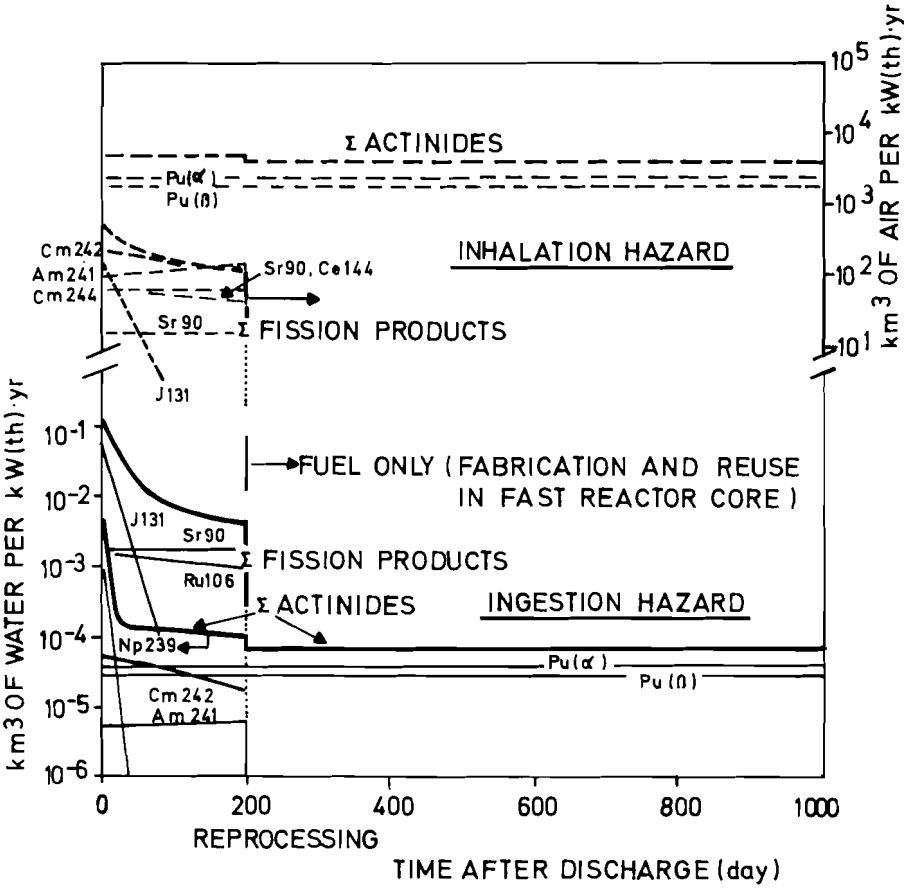

Figure V-8: Biological Hazard Potential of LMFBR Fuel

The situation of high-level waste from the fuel is summarized in Table V-XI and Figures V-9 and V-10. High-level waste is of major concern as it contains all of the fission products, Am and Cm, and approximately one per cent of U/Pu. This type of *liquid* waste is stored for five years in steel tanks for cooling. Then the high-level waste is vitrified and, depending on the national waste management scheme, the glass blocks are stored in air- and/or water-cooled buildings for approximately another 50 years ("retrievable surface storage"). Then the glass blocks are transported to the final storage (geologic formations such as bedded salt or salt domes). From this very moment, the only conceivable way by which waste could reach the biosphere is the leaching of the waste by water, which

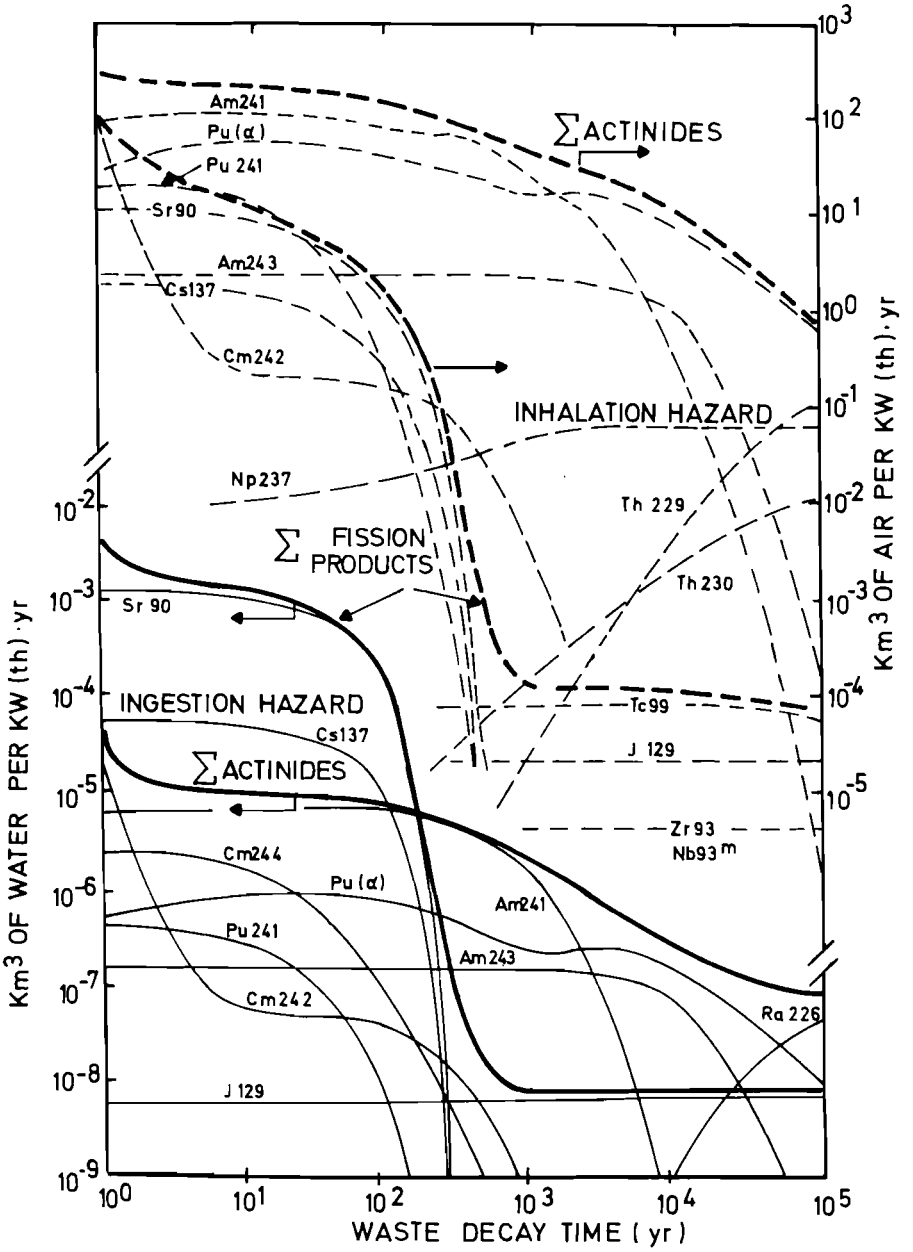

Figure V-9: Biological Hazard Potential of LMFBR High Level Waste

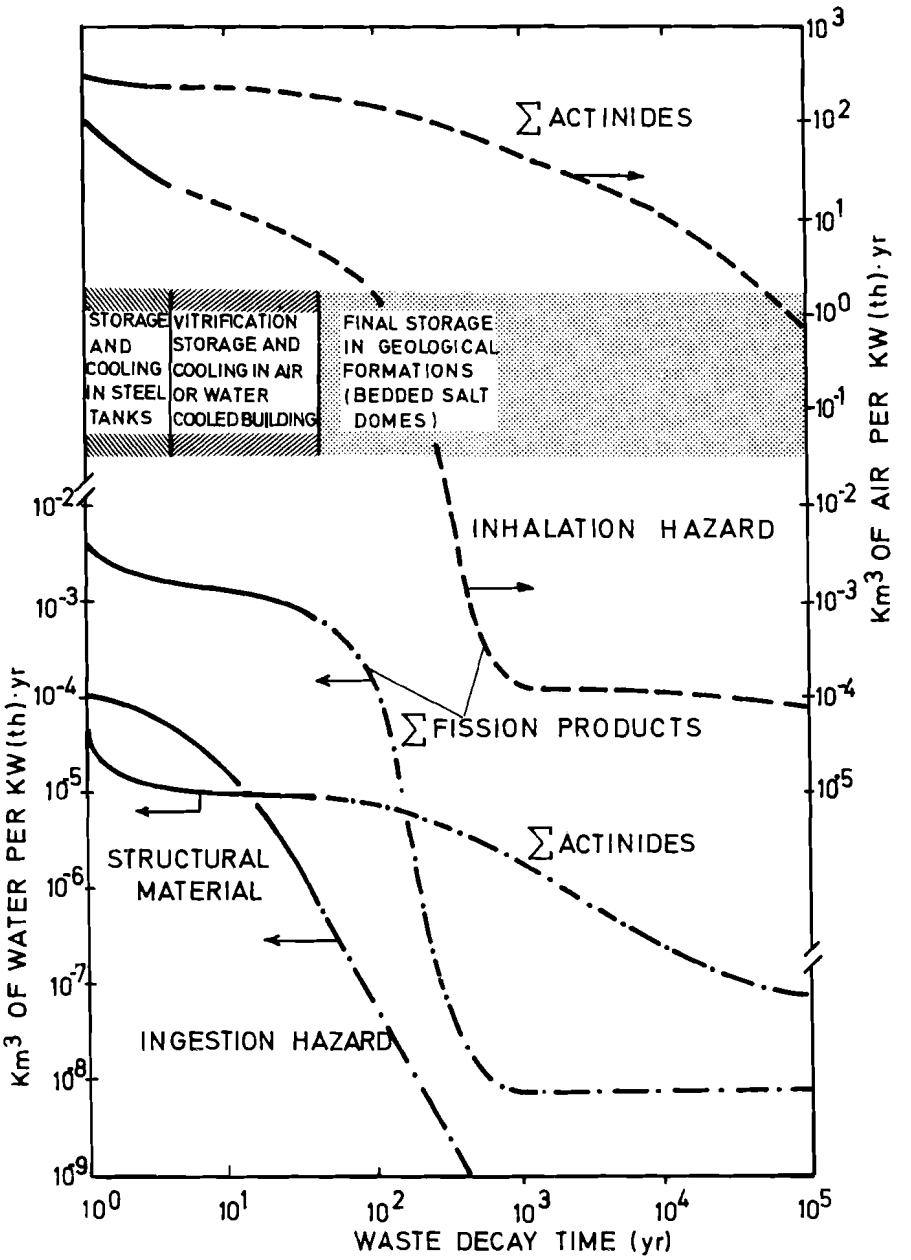

Figure V-10: Biological Hazard Potential of LMFBR High Level Waste (schematically)

penetrates into the geologic formation chosen for final disposal. By applying the approach to high-level waste one must realize that the *ingestion hazard* in terms of km^3 of water per kW(th)·yr is the more approriate indicator for the biological hazard potential of the waste ("older" than about 50 years). For any final conclusions, however, a complete accident analysis, including possibilities of the water reaching the stored vitrified waste and leaching factors, would have to be made.

We again find that, depending on the type of dilutent, either the actinides or the fission products can be dominant. For the case of an inhalation hazard (probably valid for only the first 50 years of storage), the BHP of the actinides starts out at roughly 400 km^3 of air per kW(th)·yr one year after shut-down, and decays to roughly half that value within 50 years (the time of final imbedding in geologic formations), and to roughly one km^3 of air per kW(th)·yr after 100,000 years. The fission products decay to approximately four km^3 of air per kW(th)·yr 50 years after discharge, and to approximately 10^{-4} km^3 of air per kW(th)·yr after 100,000 years.

The ingestion hazard is initially dominated by the fission products at approximately $4 \cdot 10^{-3}$ km^3 of water per kW(th)·yr immediately after reprocessing, and decays to roughly 10^{-3} km^3 of water per kW(th)·yr after 50 years, and to 10^{-8} km^3 of water per kW(th)·yr 1000 years after discharge. The BHP (water) of the fission products is then constant for the next 200,000 years because of the J129 activity. The actinides (including around one per cent of Pu isotopes) start out at an ingestion hazard of a factor of a hundred lower than for the fission products, but, due to their longer half-lives, the actinides become dominant 200 years after discharge. Thereafter, the BHP value of the actinides is one to three orders of magnitude higher than that of the fission products.

2.4 Radioactive Waste Management

During reprocessing, the fission products as well as Am, Np, Cm, and one per cent Pu/U (that is $1.22 \cdot 10^8$ Ci β/γ-activity, and $1.68 \cdot 10^5$ Ci α-activity per GW(e)·yr) are separated from the high burn-up fuel and go to the high-level liquid waste. The chemical processes applied in the reprocessing facility also increase the mass of contaminated waste considerably.

Radioactive waste can be defined by different waste categories, and various categories have been proposed. Of all the definitions, at least one is generally accepted:

- high-level waste (HLW): requires cooling by forced convection, and heavy shielding during transportation;

- medium-level waste (MLW): shielding during transportation and handling is necessary, but no cooling by forced convection (the dose rate at the surface of the drums used for disposal is greater than 200 mrem/h);
- low-level waste (LLW): no shielding, the dose rate is less than 200 mrem/h.

These three categories can also be characterized by radioactive concentration [V-8]:

high-level waste (HLW) : >10 Ci/l or Ci/kg;

medium-level waste (MLW): 10^{-4} to 10 Ci/l or Ci/kg;

low-level waste (LLW) : 10^{-8} to10^{-5} Ci/l or Ci/kg.

N.L. Franklin summarized the 1975 status of reprocessing and waste management [V-9].

R. Gasteiger [V-10] reported the following masses of waste per ton of fuel (see Table V-XII). Under the assumption of a steadily improving waste management technology, the lower limits of R. Gasteiger's data were considered reasonable for future volumes of waste. Reduction of waste volume is one of the main tasks of waste R&D programs, as could be clearly concluded from the IAEA Waste Management Conference in Vienna in 1976.

A few numbers for LMFBR waste volumes (and also for fuel fabrication) could be found in [V-11], but in general the data base of LMFBR waste volumes is not entirely satisfactory at the present time.

Taking into account the masses of Figure V-5 and adding numbers for packaged solid low- and medium-level waste volumes for the reactor itself (30 m^3/GW(e)·yr, [V-11]), which agree with numbers mentioned in the US LMFBR Environmental Impact Statement [V-12], the total amount of waste per one GW(e)·yr is such as is listed in Table V-XIII and Figure V-11.

Table V-XII: Waste Volumes

Plant	Waste Category	Waste Volume ($m^3/t_{heavy\ metal}$)
Reprocessing	High-level liquid waste	0.6
	This can be concentrated and solidified by vitrification [V-9], which leads to	0.08
	High-level solid waste (cladding hulls)	0.3
	Medium-level liquid waste	10
	This can be concentrated to MLW (liquid)	3
	MLW (organic)	2
	Both lead, after further concentration and bituminization, to [V-9]	1.5
	Medium-level solid waste	3
	Low-level liquid waste	35
	Low-level solid waste	10
Fuel Fabrication	Low-level solid waste	10

2.5 Radioactive Waste Disposal

A considerable effort is now made in a number of countries to study the long-term storage of highly active waste after solidification, vitrification, and packaging. The following alternatives are being considered:

- storage under water in ponds;
- storage in air-cooled concrete vaults;
- storage in underground caverns, or holes drilled in suitable geologic formations;
- storage of individual packages in flasks of special design which may be spaced in an array in a desert region.

Table V-XIII: Annual Quantities of Radioactive Wastes from an 1000 MW(e) LMFBR

Site of Waste Production	Waste Category	Waste Volume (m^3/GW(e)·yr) initial	final
Reprocessing Plant	HLW liquid	14	
	After concentration:		
	HLW vitrified		2
	Cladding hulls		7
	MLW liquid	240	
	After concentration and bituminization:		12
	MLW solid		70
	LLW liquid		830
	LLW solid		240
Fabrication Plant	LLW solid, solidified		230
Reactor	MLW solid, solidified		30
	LLW solid, solidified		
Whole Fuel Cycle	LLW, MLW solid		≈600
	HLW solid (glas , hulls)		≈10
	LLW liquid		≈800

In all these cases, the waste would be in a fully retrievable form and some surveillance would be required.

In addition, all countries with major nuclear programs are already investigating possible methods for the ultimate disposal of solidified, vitrified, and packaged waste with the object of completely eliminating any future need for surveillance. A very wide range of possibilities has been considered:

- disposal into caverns, tunnels, or holes drilled in geologic formations of hard rock, salt, or clay (Figure V-12 shows a possible disposal in a salt cavern);
- disposal to the deep ocean or holes drilled into the seabed.

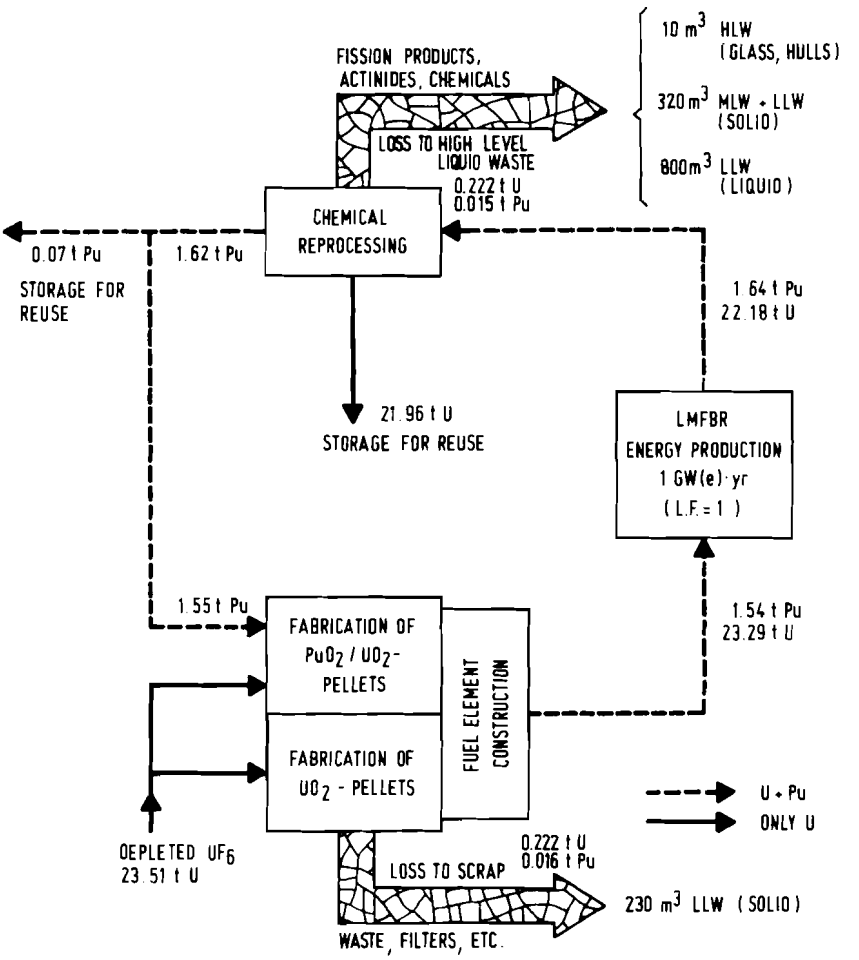

Figure V-11: Integrated LMFBR Fuel Cycle System--Yearly Mass Flows of Uranium, Plutonium, and Radioactive Waste for One GW(e)

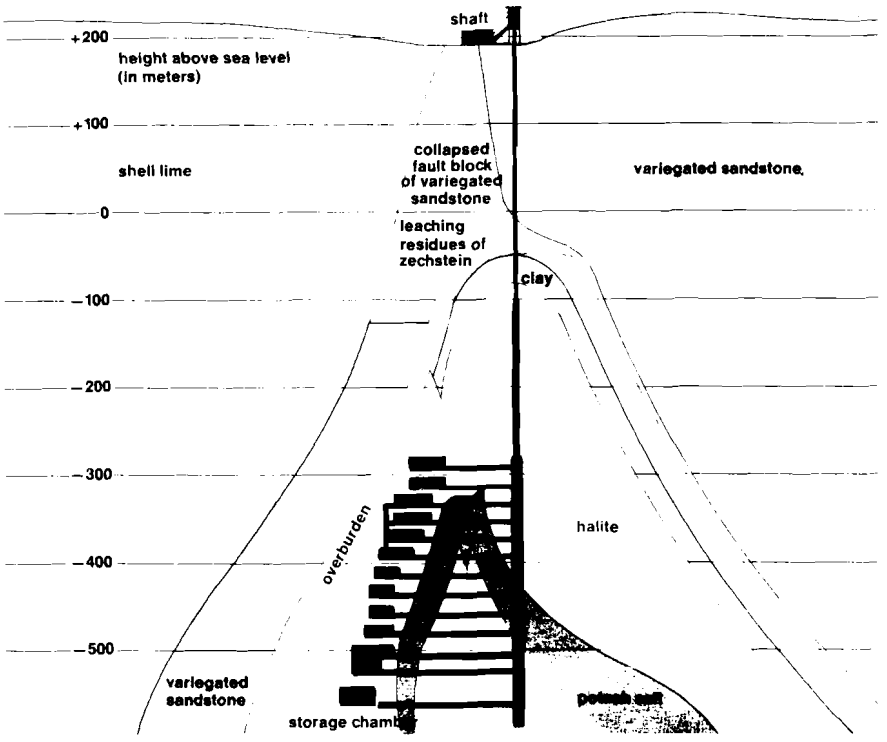

Figure V-12: Waste Storage in Salt Mines

Figure V-13 exemplifies the ingestion hazard curves for high-level waste, shown already in Figure V-10, as compared to the hazard potential of uranium ore. The hazard potential for uranium ore was defined as corresponding to 16,450 t of uranium ore which (according to J. Hamstra [V-13]) must go through uranium milling and enrichment facilities so that 23 t of U with 0.2 per cent U235 content is left in the tail end of the enrichment plant. The 23 t of uranium is loaded into the LMFBR reactor core and ultimately converted into energy, having passed many times through the LMFBR fuel cycle (see Figure V-5). The hazard potential of the 16,450 t uranium ore appears to be an appropriate basis for comparison, since it allows one to compare the initial uranium-ore hazard potential in existing geologic sites to the hazard potential of waste generated per year of power production in an LMFBR.

The waste-disposal technique largely decreases the BHP per unit volume by dilution in non-radioactive geologic layers (salt domes, etc.). By this procedure one obtains *average potential factors* that are similar to those of uranium ores.

It has been stressed already that the BHP index is only a better base for comparing different nuclear inventories, compared to the sole utilization of activities in terms of curies. It represents the volume of water or air that would be needed to dilute the material down to the maximal permissible concentration, assuming that the isotopes are uniformly

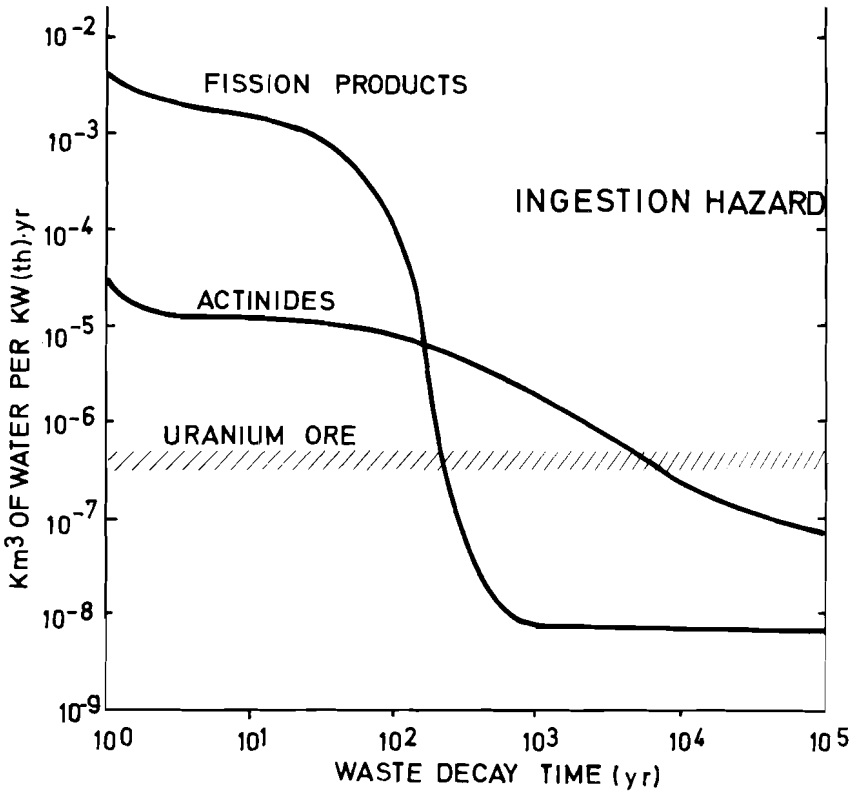

Figure V-13: Biological Hazard Potential of LMFBR High Level Waste as Compared to Uranium

mixed with dilutent. But the hazard index approach must not be overinterpreted. As illustrated in Figure V-13, after a few thousand years the BHP index of the waste is lower than that of the ore. But it would be completely misleading to infer from that diagram that waste represents a lower risk than the ore from which it came. The BHP index approach cannot serve as a substitute for complete risk analysis. For example, it does not make allowance for the fact that the toxic components of the ore, which are immobilized in solid rock, have been made much more liable to escape into the biosphere by mining and milling of the ore.

Long-term disposal methods will require very careful study, development, and demonstration to ensure that they will effectively isolate the waste from man and his environment.

Until the investigations of ultimate long-term disposal without surveillance will have been completed, the chief methods used will be the long-term storage techniques which require some surveillance and which are available now. As will be explained in Chapter X, this may well be so until the year 2000, since up to that time no major waste volumes will be generated by the slowly growing fast-breeder economy.

2.6 Integrated Biological Hazard Potentials of the LMFBR

The integrated BHP (IBHP)* is one way of incorporating the fact that some materials with a long physical half-life could pass through several generations, each with its own biological half-life. We have calculated the IBHP for the reference LMFBR fuel cycle, and the results are summarized in Table V-XIV. Only the results for water are quoted here because of the solidified (glassified) form that the material is in after reprocessing. A few major points from Table V-XIV are worth quoting:

(1) The total IBHP (water) for the LMFBR reference case is approximately 88 per cent due to the fission products, and the main components of that amount are I129 and Sr90;

(2) The IBHP (water) for the actinides is approximately 12 per cent of the total for the LMFBR cycle;

(3) The cladding contributes very little to the long-term hazard index, compared to the waste from the fuel.

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* See page 233 for explanation of terms.

Table V-XIV: Summary of Integrated Biological Hazard Potentials for the LMFBR Waste Cycle[a]

Isotopes	$t_{\frac{1/2}{0.693}}$ (sec)	Activity Inventory[b] (Ci/kW(th)·yr)	MPC (Ci/km^3_{water})	IBHP (km^3_{water}·sec/ /kW(th)·yr)
SS 316 Cladding				
Mn53	$8.6 \cdot 10^{13}$	$3 \cdot 10^{-7}$	$3 \cdot 10^{3}$	$8.6 \cdot 10^{3}$
Mn54	$3.8 \cdot 10^{8}$	3.2	$1 \cdot 10^{5}$	$1.2 \cdot 10^{4}$
Total				$2.1 \cdot 10^{4}$
Fission Products				
Sr90	$1.3 \cdot 10^{9}$	0.30	300	$1.3 \cdot 10^{6}$
Cs137	$1.4 \cdot 10^{9}$	1.33	$2 \cdot 10^{4}$	$9.3 \cdot 10^{4}$
I129	$7.7 \cdot 10^{14}$	$3.1 \cdot 10^{-7}$	60	$4.0 \cdot 10^{6}$
Tc99	$9.6 \cdot 10^{12}$	$1.9 \cdot 10^{-4}$	$2 \cdot 10^{5}$	$9.1 \cdot 10^{3}$
Total				$5.3 \cdot 10^{6}$
Actinides				
Pu239[c]	$1.1 \cdot 10^{12}$	$3.3 \cdot 10^{-4}$	$5 \cdot 10^{3}$	$7.2 \cdot 10^{4}$
Pu240[c]	$3 \cdot 10^{11}$	$5 \cdot 10^{-4}$	$5 \cdot 10^{3}$	$3 \cdot 10^{4}$
Pu242[c]	$1.8 \cdot 10^{13}$	$1.5 \cdot 10^{-6}$	$5 \cdot 10^{3}$	$5.4 \cdot 10^{3}$
Cm242	$7.3 \cdot 10^{9}$	1.1	$2 \cdot 10^{4}$	$4.0 \cdot 10^{5}$
Am241	$2 \cdot 10^{10}$	$2.5 \cdot 10^{-2}$	$4 \cdot 10^{3}$	$1.3 \cdot 10^{5}$
Am243	$3.4 \cdot 10^{11}$	7.4	$4 \cdot 10^{3}$	$6 \cdot 10^{4}$
Total				$7 \cdot 10^{5}$
Total				$6.0 \cdot 10^{6}$

[a] Only the major isotopes are quoted here for the sake of brevity;

[b] At t = 0 (time of discharge);

[c] Only one per cent of discharge amount.

3. RADIOACTIVE INVENTORIES OF FUSION REACTORS

There are at least ten sources of radioisotopes in D-T, D-D, and even D-He3 fusion reactors:

(1) tritium in the breeding medium, vacuum pumps, fuel reprocessing system, or that which is held in reserve;

(2) activated metallic structural components in the reactor blanket;

(3) blanket coolant including impurities;

(4) breeding medium and its associated impurities;

(5) neutron multiplier (if required) and impurities in it;

(6) shield material including structure, coolant, neutron and gamma ray absorbers;

(7) magnet structures and supports, or laser optical systems;

(8) air or other gaseous environment around the reactor;

(9) auxiliary equipment in close proximity to the reactor, such as neutral beam injectors, vacuum pumps, control devices, etc;

(10) biological shielding (usually concrete) and building structure.

We will now attempt to put the entire radiological picture for fusion reactors into perspective, and first list the potential amounts of radioactivity which could be present in each of these ten categories. Next, the question of the biological hazard potential will be addressed, using standard maximum permissible concentrations of the various isotopes in the biosphere. Finally, the probability that the radioisotopes could find their way into the biosphere will be explored in Chapters VI to VIII.

Most of the treatments in the past have concentrated on items *(1)* and *(2)* in the foregoing tabulation, as these were clearly dominating in early D-T reactor designs. However, as proposals for reducing the tritium inventory in the blanket and the activation of the blanket structural materials have been made, the residual radioactivity in *(3)* to *(10)* may become more important. Unfortunately, it is impossible at this stage of fusion reactor design to address the problems in *(9)* and the laser part of *(7)*. We also do not have sufficient information about most reactor designs to be completely quantitative with respect to *(4)*, *(5)*, *(8)*, and *(10)*.

3.1 Tritium Inventory

We will consider the case of the D-T fuel cycle first, and make a few comments on how this would be modified by the D-D or D-He3 cycles later. The major sources of tritium in a D-T fusion reactor are listed below:

(a) fuel-injection system (pellets or gas);

(b) energetic neutral-beam heating (if required);

(c) plasma;

(d) vacuum system and unburnt-fuel extraction system;

(e) tritium implanted in the first walls;

(f) tritium purification system (separation of H, D, T, and He isotopes);

(g) blanket breeding medium;

(h) tritium extraction system for breeding system;

(i) blanket structural material (mainly due to solubility of hydrogen isotopes in metals);

(j) blanket coolant (if exclusive of *(g)*);

(k) tritium extraction system for blanket coolant (if exclusive of *(h)*);

(l) power system interfacing with blanket coolant (i.e. heat exchangers, secondary coolants, steam cycle, etc.);

(m) magnet shield (if it contains boron);

(n) magnet shield coolant and associated power system;

(o) building ventilation system;

(p) storage for start-up or unexpected break-down in tritium processing system.

The total inventory depends on many parameters such as

- reactor type (TOKAMAKs may require divertors, Mirrors may have direct convertors, Theta Pinches may have cold-gas blankets, lasers will have fuel-pellet fabrication facilities, etc.);
- structural material (the solubility and permeability of tritium is quite sensitive to materials);
- breeding medium (liquid-lithium systems may have higher tritium inventories than solid-lithium compound breeders);
- plasma volume and fractional burn-up per pass (mainly determines the amount of tritium that must be handled, as burn-ups can be as low as one per cent in TOKAMAKs or as high as 30 per cent in laser systems);
- coolant;
- power cycle (steam, direct-cycle turbine, etc.).

A complete treatment of all of these topics is obviously beyond the scope of this paper, but if we confine our attention to a D-T TOKAMAK reference reactor, we can significantly reduce the number of important parameters. For example, we can essentially neglect items *(c)*, *(e)*, *(i)*, *(j)*, *(k)*, *(l)*, *(m)*, *(n)*, and *(o)*. The tritium contained in each one of these systems is typically less than one gram (10^4 Ci) per 1000 MW(th) at any given time.

Another condensation can be made with respect to the fueling system by combining items *(a)*, *(b)*, *(d)*, and *(f)*. Since this is a closed loop, we can address the inventory in terms of how long it takes a non-reacting tritium atom to complete the circuit from injection, collection, and separation to reinjection. It will be shown later that this *must* be on the order of one day or less, if reasonable tritium inventories are to be maintained. The above analysis then leaves us with three main systems:

- tritium injection, collection, and purification system;
- blanket breeding medium, and its extraction system;
- storage system.

The estimate of the amount of tritium in the fueling system must begin with the fact that it takes roughly 0.14 kg of tritium (in conjunction with the appropriate amount of deuterium) to produce 1000 MW(th)·day of energy. However, the fueling system must handle much more tritium than this, because only a small fraction of this is burned up per pass [V-14 to V-16]. If we choose an average value of one per cent [V-17], then approximately 14 kg of tritium per day must be injected into the plasma per 1000 MW(th) generated. This translates into roughly 60 kg/day for a 5000 MW(th) reference system. It is easy to see from this number that, if it takes as long as ten days for a non-reacting tritium atom to pass through this circuit, the tritium inventory in the fueling system is in excess of 600 kg, a completely unacceptable value. On the other hand, if one could process the tritium in 0.1 day, the inventory would be on the order of approximately 6 kg. Current estimates of the cycle time (e.g. for cryopumps) are more like six hours, and we will use that for our reference case.

The inventory of tritium in the breeding medium depends mainly on whether liquid lithium or solid lithium-containing compounds are used. If one can extract tritium from high-temperature lithium at the 5 appm (atomic ppm) level, inventories on the order of 2 kg/1000 MW(th) are expected [V-14, V-17]. If one uses solid breeders such as LiAl, LiO_2, or $LiAlO_2$, the inventory in the blanket could be as low as 25 g/1000 MW(th) [V-15, V-18].

Unfortunately, these numbers are probably too low for the ceramic compounds, because the calculations assume that one can maintain small (approximately 10 micron diameter) particles for long times (two to three years) at high temperatures (800 to 1100°C) and under neutron bombardment. Sintering will undoubtedly take place, causing the average particle size to increase dramatically. The problem can be better understood by noting that the inventory goes as

$$\text{Inventory} \propto \frac{(T_2 \text{ Generation Rate}) \cdot (\text{Radius of Breeder Particle})^2}{(\text{Diffusity of Tritium in the Breeder})}$$

It is conceivable that the lithium ceramic particle sizes could increase to 1/10 mm or more in diameter, similar to the sintering observed in UO_2 fuel rods, in which case solid breeder inventories could rise to the order of 1 kg/1000 MW(th). Recent calculations [V-19] have revealed that in the UWMAK-II design [V-15] the tritium inventory could be as much as 0.5 kg due to simple temperature gradients alone, which are inherent to ceramic materials.

One must also include the amount of tritium in the extraction circuit for the breeder material. This has been estimated as ranging from 0.1 kg/1000 MW(th) for liquid Li [V-20], to 0.003 to 0.08 kg/1000 MW(th) for solid breeder systems [V-17, V-18]. Taking the more pessimistic numbers, this would translate into 0.5 kg for our 5000 MW(th) reference TOKAMAK system, or 0.1 kg/1000 MW(th). It is expected that future D-T fusion power plants will have to run with a sufficient reserve of tritium to compensate for malfunctions in the tritium-processing systems (or provide for duplicate facilities). For example, it would be very expensive to shut down the entire plant if only the tritium-separation unit failed. The time to repair such equipment is difficult to estimate, but since it will undoubtedly be contaiminated, one certainly could envision hours or days for even simple repairs. This would require approximately 0.1 kg/1000 MW(th)·day of makeup tritium for the repair of the extraction system for the breeding material. A more serious problem would be the disruption of the fuel-handling system where, as we stated before, the throughput might approach 14 kg/day per 1000 MW(th). Clearly, one would want to have on the order of one third to one day of tritium on reserve, and we might estimate a value of roughly five to 14 kg per 1000 MW(th) as a reasonable possibility.

This brief analysis then produces the following tritium inventory numbers for a TOKAMAK reactor:

	Tritium Inventory (kg/1000 MW(th))
Tritium injectors, collectors, and purification system	3*
Breeding medium, and separation unit	0.5 - 2
Storage and emergency inventory (0.3 to 1 day)	5 - 14
Total	8 - 19

For the purpose of this study we will proceed with the analysis where roughly 10 kg tritium is required per 1000 MW(th), fully recognizing that slightly different designs might reduce that number, or that reduced burn-up values might drastically increase it. An important point of this analysis is the fact that the use of solid breeders will *not* significantly reduce the total tritium inventory in a reactor. However, the use of solid breeders would reduce the amount of tritium in contact with high-temperature systems, and hence reduce the amount which has the greatest potential for release. The storage of tritium can be accomplished quite easily and safely in areas well removed from the reactor. There is a much higher chance of tritium being released at any one of the steps in the fuel handling system, especially since every effort will be made to handle tritium quickly in order to reduce the overall inventory.

The general topic of potential for tritium release and its effect on humans will be treated in more detail later (see Chapter VI).

3.2 Radioactivity in Metallic Structural Components

In order to understand this problem and the ways in which it can be ameliorated, one must go back to the fundamental features of a D-T fusion reactor. Most of the energy (80 per cent) of the fusion reaction is carried away by the 14 MeV neutrons, and the object of the reactor blanket is to extract this kinetic energy by thermalizing the neutrons while, at the same time, breeding at least one more tritium atom from lithium. In a basic sense, there is no connection between the moderation, the breeding, and the structural support features of the blanket. That is, there is no fundamental reason prohibiting the moderation and breeding from taking place *inside* the vacuum chamber. Furthermore, the reactor structural components (those that must provide the framework to contain the coolant pressures,

* See third paragraph on page 221.

vacuum loads, and magnetic forces) can be protected from even the thermalized D-T neutrons if boron-containing compounds are also placed into the vacuum chamber.

Such a scheme has indeed been proposed, using graphite to moderate the neutrons, high-temperature lithium ceramic compounds for breeding, beryllium as a neutron multiplier, and boron compounds to absorb thermal neutron [V-20 to V-22]. This scheme relies on passive radiative cooling of the energy conversion and breeding blanket, and diffusion of the tritium into the vacuum chamber to be collected with the "unburnt" fuel.

The more conventional way of designing fusion reactor blankets is to place the moderators and breeding materials *behind* the first wall in close proximity with the coolants. Such a scheme requires the 14 MeV neutrons to pass *through* the first vacuum wall and other supporting metallic components, where they are then thermalized and used to breed more tritium. These neutron fluxes can cause considerable activation of the metallic components and the impurities contained within them.

There have been at least ten detailed studies of the activation of CTR blanket materials [V-14 to V-18, and V-23 to V-28]. Table V-XV includes data from a self-consistent study to demonstrate how sensitive (or insensitive) the induced activity is to the choice of the metallic structural material. We have chosen to include not only the most likely material to be used for near-term fusion power plants (austenitic stainless steel) but also more exotic refractory alloys for high-temperature application (a V-Ti alloy, Nb-1Zr, and a molybdenum-based alloy TZM); one is especially chosen for its low induced activity (Al 2024).

One must be careful at this point to emphasize the difficulties associated with refractory metals and aluminum in relation to a D-T fusion-reactor environment. For example, aluminum alloys must operate below approximately 200^{0}C to minimize creep and helium-gas embrittlement problems [V-29]. Aluminum is particularly susceptible to high helium-gas generation in a 14 MeV neutron environment because of its large (n,α) cross-section (approximately 120 mb). This can result in as much as 400 appm of helium per year per MW/m^2 wall loading [V-30]. Severe embrittlement can occur at levels one-tenth of that value when alloys are operated above half of the homologous temperature [V-29]. Added to these problems is the incompatibility of Al with liquid lithium (thus forcing the use of solid breeders). Clever designs have been able to partially offset the low-temperature problems by cooling the first wall with water (absorbing one-third of the reactor heat), and then lining the Al walls with a thermal insulator (Al_2O_3, or graphite felts), so that very high-temperature helium can be passed through the center of the cell to absorb the other two-thirds of the heat [V-31]. Unfortunately, this also raises questions with respect to contamination of coolant water with tritium,

Table V-XV: Comparison of Induced Activity for Several Potential CTR Structural Materials in a Common Blanket Design (at shut-down after 2 year operation)[a] (Ci/kW(th))

Isotope	Half-life		SS 316[b]	TZM[c]	V-Ti[d]	Al 2024[e]	Nb-1Zr[f]
Na24	14.96	h	-	-		256	
Mg27	9.46	min	0.036	-		212	
Al26	$7.4 \cdot 10^5$	yr	-			$9 \cdot 10^{-5}$	
Al28	2.3	min	13.1	-		118	
Ca45	165	day	0.01	0.26	4.8		
Sc46	83.9	day	0.01	0.47	8.7		
Sc47	3.43	day	0.01	0.48	8.8		
Sc48	1.83	day	0.02	0.98	38.4		
Sc49	58	min	-	0.05	0.91		
Ti45	3.09	h	-	0.02	0.43		
Ti51	5.8	min	0.14	-	79.3		
V 49	330	day	1.17	-	1.69	0.002	
V 52	3.75	min	35.4	-	1026	0.37	
Cr51	27.8	day	100	-			
Mn53	$1.9 \cdot 10^6$	yr	$5 \cdot 10^{-6}$	-		$1.8 \cdot 10^{-8}$	
Mn54	303	day	53.5			4.44	
Mn56	2.58	h	352			25	
Mn57	7	day	2.43				
Fe55	2.6	yr	195			0.63	
Fe59	45.6	day	0.14				
Co57	270	day	17.4				
Co58	71.3	day	90				
Co60m	10.5	min	10.4				
Co60	5.26	yr	4.8			0.3	
Ni57	36	h	3.54				
Ni59	$8 \cdot 10^4$	yr	$3 \cdot 10^{-5}$				
Ni63	92	yr	0.036				
Cu64	12.8	h	-	-	-	148	
Zn65	245	day	-	-	-		
Sr89	52.7	day	0.002				0.02
Sr90	27.7	yr	$4 \cdot 10^{-6}$				$4 \cdot 10^{-5}$
Y 90	64	h	-	0.02			11.6
Y 91	58.8	day	-	-			0.06
Zr89	78.4	h	0.18	8.97			2.74
Zr95	65.5	day	0.04	2.21			
Nb92m	10.2	day	0.42	21.1			717[g]
Nb94m	6.3	min	-	-			4390[h]
Nb94	$2 \cdot 10^4$	yr	-	-			$6 \cdot 10^{-4}$
Nb95m	90	h	0.07	3.49			16.2
Nb95	35	day	0.21	6.87			16.6

Table V-XV: Comparison of Induced Activity for Several Potential CTR Structural Materials in a Common Blanket Design (at shut-down after 2 year operation)[a] (cont'd) (Ci/kW(th)

Isotope	Half-life	SS 316[b]	TZM[c]	V-Ti[d]	Al 2024[e]	Nb-1Zr[f]
Nb96	23.4 h	0.09	4.59			
Nb97	72 min	0.065	2.29			
Mo91	15.5 min	1.28	65.3			
Mo93	10,000 yr	$5 \cdot 10^{-3}$	0.24			
Mo99	66.7 h	28.7	1507			
Mo101	14.6 min	8.3	439			
Tc99m	6.87 h	28.7	1507			
Tc99	$2.15 \cdot 10^{5}$ yr	2.10^{-4}	10^{-2}			
Tc101	14 min	8.3	451			
Total		1062	4120	1261	884	5155

[a] Blanket design is a toroidal blanket, of major radius 1300 cm, plasma radius 500 cm, first-wall radius 550 cm, 2823 m^2 area, 0.4 cm thick first metallic wall, 51 cm of 95% Li and 5% metal, 15 cm of reflector, 5 cm of 95% Li and 5% metal, and 2 cm of metal wall. Power level of reactor is 5025 MW(th) and neutron wall loading 1.25 MW/m^2 [V-28];

[b] 63% Fe, 19% Cr, 12% Ni, 2% Mn, 2% Mo, 2% Si (atomic)

[c] 98.9% Mo, 1% Ti, 0.1% Zr (atomic);

[d] 84% V, 16% Ti (atomic);

[e] 94.51% Al, 3.02% Mg, 0.48% Si, 0.046% Cr, 0.45% Mn, 0.24% Fe, 2.15% Cu, 0.098% Zn (atomic);

[f] 99.25% Nb, 0.75 Zr (atomic);

[g] 50% isomeric ratio to Nb92m;

[h] 50% isomeric ratio to Nb94m.

sintering of solid breeders, and the requirement of using large amounts of the scarce element Be for neutron multiplication*. The high frequency of changing the first

* There have been proposals for using Pb as neutron multiplier, but no self-consistent design has been made to demonstrate adequate shielding of magnets and adequate breeding ratios when appropriate extra structure has been added, neutron leakage accounted for, and dimensional stability of the breeder assured.

wall because of radiation damage would also tend to reduce the plant factor, and hence increase costs. Even with all of these problems, it is still worthwhile to calculate the induced activity in Al alloys, if only to illustrate the absolute best one can do with the available metals* .

Before we get into the discussion of radioactive inventories we should also make a few points about the use of large amounts of refractory alloys for fusion application. First of all, there is presently no well established industry (or mining capacity) that could supply the necessary tonnage of refractory (Nb, Mo, V) metals (in million ton quantities) required for a well-established fusion economy. The alloys which have been proposed, such as V-20Ti, are truly in the experimental stage with little or no commercial history, let alone irradiation history. Fabrication and cost problems are considerable, as is the problem of low ductility of the finished product. Nb-1Zr is somewhat better with regard to commercialization, but very little is known about its irradiation behavior. It also suffers more from availability and cost disadvantages. Molybdenum alloys appear to represent less of a burden on resources, and can perform respectably in a CTR environemnt. Unfortunately, joining techniques for Mo alloys are difficult to implement [V-17].

Stainless steel has been studied for over 15 years for nuclear applications, and a mature industry now exists to manufacture million ton quantities under strict quality-assurance standards. It also has problems with lithium compatibility above 500^{0}C [V-14] and the use of scarce alloying elements, although the differences between an economy with and without fusion reactors is not overwhelming for the world.

The above statements are meant to illustrate that one cannot simply solve the reactor-activation problem by choosing some non-activating element from the periodic table without regard for the performance of the blanket. There are no easy and obvious choices, but if one were to be made today, it would probably be to use some form of austenitic-steel alloy. Nevertheless, we will consider all of the above alloys for potential CTR fusion-reactor application to illustrate some degree of flexibility.

Returning to Table V-XV, we have given the activation of the various isotopes in Ci/kW(th) for the five alloys stated previously. The same blanket design (similar to that described in [V-14]) and volumetric proportions were used for all of the calculations. Basically, it is a 73 cm thick configuration,

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* We will not discuss non-metal blanket structures here. The use of C and SiC has been proposed [V-32], but the vacuum integrity of such systems needs to be demonstrated before they can be seriously considered.

which has a 0.4 cm thick first wall, followed by a 95 per cent natural Li and 5 per cent structure region 51 cm thick, a 15 cm thick reflector, and a 5 cm zone of Li (95 per cent) and structure (5 per cent), and a 2 cm thick outer shell. Slight modifications would have to be made to adjust for the strength per unit volume of material at the anticipated operating temperatures and, of couse, it would be difficult to use Li with Al alloys. Nevertheless, keeping a constant design permits a first-order comparison.

An important feature of the induced radioactivity of CTR structural components is the rate at which it is built up. This is illustrated in Figure V-14, where the normalized radioactivity level in the test blanket described above is given as a function of irradiation time. The induced radioactivity after only 1000 sec of operation ranges from 10 per cent (in the case of TZM and steel) to 93 per cent (for Nb-1Zr) of the value after 30 years of operation. After one day of operation, at least 50 per cent of the equilibrium induced activity is achieved, and by ten days it is over 90 per cent in all metals

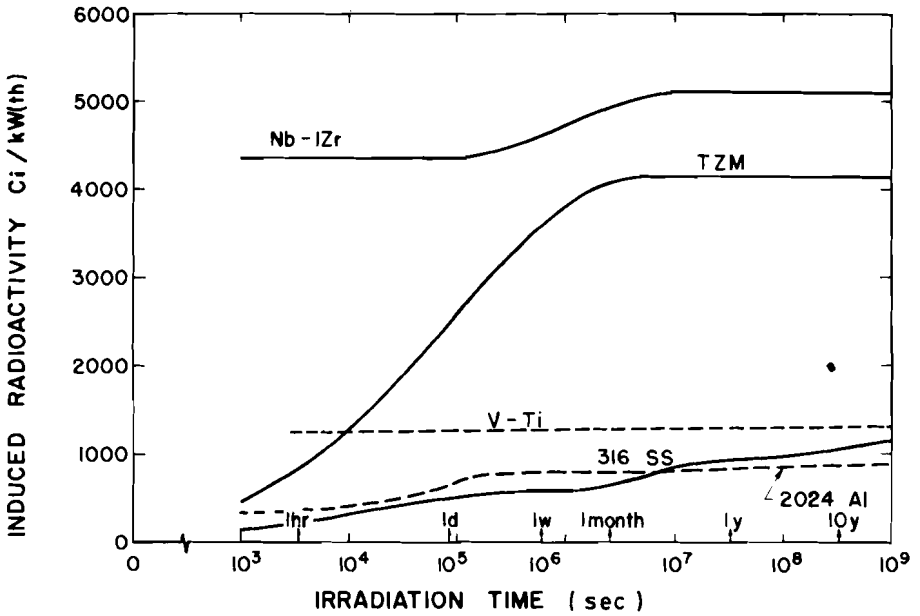

Figure V-14: Effect of Reactor Operation Time on Induced Activity in CTR Structural Materials

except the steels. The significance of these observations is that fusion reactor structures will be very radioactive almost immediately after start-up operations. Another important feature of Figure V-14 and Table V-XV is to note that the relative activity after two years of operation differs by only a factor of six from the aluminum alloy (Al 2024)* to the Nb-1Zr alloy.

The next point to consider is the rate at which this induced activity decays away after the reactor is shut down. Figure V-15 shows the activity as a function of time after a two-year operation cycle.

There are several important features of this curve. First, in the short time after reactor shut-down (approximately 10^0 to 10^5 sec) which might be pertinent to an accident case, the radioactivity levels of all five materials are within a factor of 50 of each other, the vanadium alloys the lowest, and molybdenum alloys the highest. The next time period of interest is the 10^5 to 10^6 sec range where maintenance might be performed. Here the Al alloys have the lowest radioactivity, being about a factor of 100 lower than Nb-1Zr and SS 316, and a factor of about 30 lower than the TZM. The final period of interest is the long-term (longer than 100 years) storage regime, where V-Ti alloys have a clear advantage of no long-lived isotopes. Mo93 (half-life: 10,000 years) causes the radioactivity in TZM to temporarily level out at the 0.2 Ci/kW(th) level, and SS 316 to approach a level of approximately 10^{-2} Ci/kW(th). Ni59 (half-life: 80,000 years) would cause the activity in SS 316 to level out again at $2 \cdot 10^{-4}$ Ci/kW(th) for times exceeding 50,000 years. Nb94 (half-life: 20,000 years) causes this alloy to reach a temporary plateau in activity at $6 \cdot 10^{-4}$ Ci/kW(th), and, finally, Al26 (half-life: 740,000 years) limits the residual activity of the Al alloy to $9 \cdot 10^{-5}$ Ci/kW(th). The times required to decay the lower radiation levels are summarized in Table V-XVI.

These numbers are put in perspective by noting that the activity per unit volume is actually quite low (see Table V-XVII). The first wall, which will have the highest level of activation, has values ranging from 27 to 160 Ci/cm^3 at shut-down. However, after 100 years of storage, the radioactivity density in the wall would be approximately the same as in natural uranium for the Al and Nb alloys, and considerably lower in the case of the V-Ti alloy. The activity per cm^3 is higher for SS 316 and TZM (by a factor of 1000) than in natural uranium, and these materials would have to be stored and monitored for longer periods of time.

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* 77 per cent of this activity is due to the Al alone, and 19 per cent is due to the copper-alloying element.

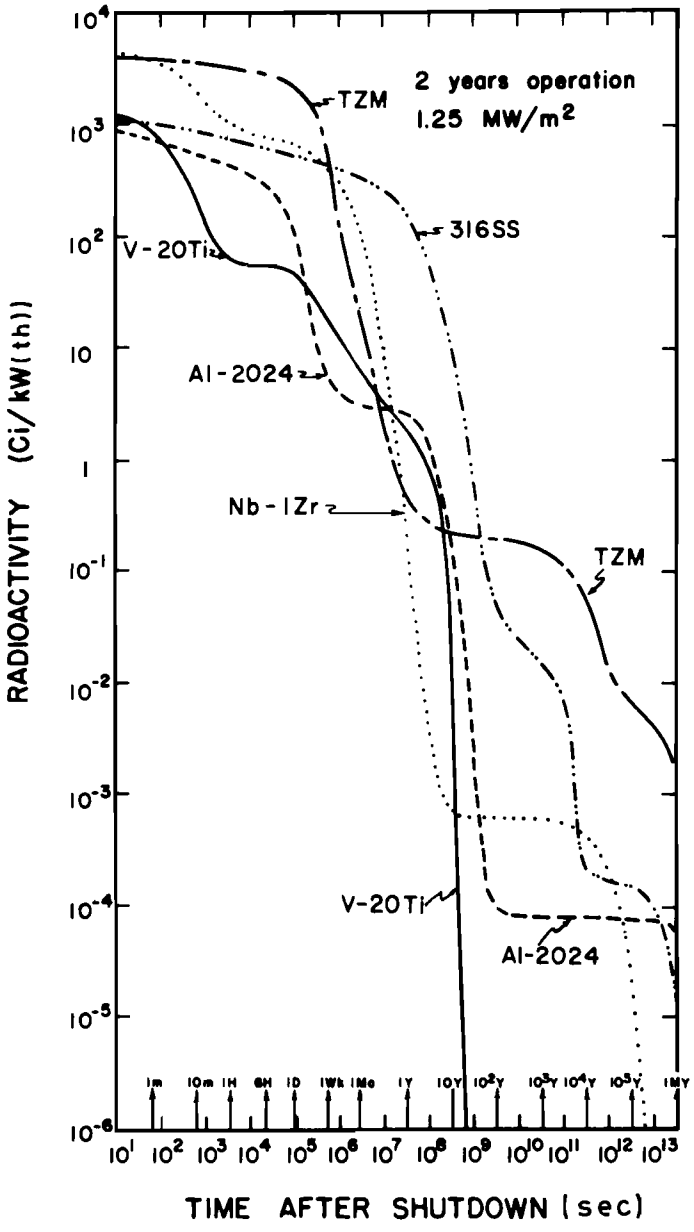

Figure V-15: Radioactivity of CTR Blankets After Shut-down

Table V-XVI: Time Required to Reach Specific Levels of Radioactivity Inventories in the Metals of Various CTR Reactor Designs--two year operation

	Approximate Time Required to Achieve Radiation Level				
	SS 316	TZM	Nb-1Zr	V-20Ti	Al 2024
Initial Level Ci/kW(th)	1060	4120	5150	1260	880
10^{3} Ci/kW(th)	100 sec	3 day	1 day	1 min	-
10^{2} Ci/kW(th)	3 yr	14 day	30 day	30 min	1 day
10^{1} Ci/kW(th)	15 yr	40 day	0.5 yr	14 day	3 day
10^{0} Ci/kW(th)	30 yr	0.5 yr	1 yr	3 yr	4 yr
10^{-1} Ci/kW(th)	60 yr	3000 yr	1.5 yr	10 yr	20 yr
10^{-2} Ci/kW(th)	2000 yr	4.10^{4} yr	2 yr	15 yr	30 yr
10^{-3} Ci/kW(th)	5000 yr	10^{6} yr	5 yr	20 yr	40 yr

Table V-XVII: Maximum Radioactivity Density in First Wall of Various CTR Blankets (Ci/cm^3)*

Alloy	Decay Time			
	t = 0	1 day	1 yr	100 yr
SS 316	100	68	29	$5 \cdot 10^{-3}$
TZM	125	83	$4 \cdot 10^{-2}$	$7 \cdot 10^{-3}$
V-20Ti	27	6.6	0.31	$<10^{-37}$
Al 2024	44	8.7	0.3	$1 \cdot 10^{-5}$
Nb-1Zr	158	94	$6 \cdot 10^{-4}$	$1 \cdot 10^{-5}$
Natural Uranium	$6 \cdot 10^{-6}$			

* Two year operating time, 1.25 MW/m^2.

The simple measurement of curies of activity in a structural component is not a very good index of the relative hazards of various radioisotopes, because it does not consider the biological effect of the decay process on humans. A somewhat better (although admittedly not perfect) approach is to consider the biological hazard potential (BHP) of an isotope, which is simply the level of radioactivity divided by the maximum permissible concentration (MPC) in the air or water. The MPC values used in this study to calculate BHP are given in Tables V-XVIII and V-XIX with the numbers of BHP for the five alloys. It must be remembered that these values refer to inventories, and not necessarily to the amount which escapes the reactor. The BHP values are quoted for both air and water, with the former more realistic in the case of accidental release, and the latter for long-term waste storage. The BHP (air) at shut-down after two years of operation varies from a low value of 27 km^3 of air per kW(th) in the V-20Ti alloys, to a high value of 391 km^3 of air per kW(th) in TZM alloys. In the case of BHP (water), the V-20Ti alloy also has the lowest value at shut-down at $4 \cdot 10^{-3}$ km^3 of water per kW(th), but the highest value is that of the Nb-1Zr alloy (0.26 km^3 of water per kW(th)).

The decay of the BHP (air) and BHP (water) factors for the various CTR alloys is shown in Figures V-16 and V-17. Rather than examining the decay behavior in detail, it is more instructive to consider the BHP (air) at short times (less than one day) for possible accidental release, and the BHP (water) for long-term (more than 100 years) waste-storage problems. The reader is urged to examine Tables V-XVIII and V-XIX and Figures V-16 and V-17 for more information on intermediate times.

Considering the short-term BHP of the alloys in air, we see that the values are roughly constant (within a factor of two) during the first day after shut-down, with the following values after one day of decay:

	BHP (air) (km^3/kW(th))
TZM	300
SS 316	200
Nb-1Zr	40
V-20T	20
Al 2024	20

Hence, the relative difference in BHP (air) for the five structural materials considered here is only a factor of about 15 for the first day after reactor shut-down.

In Table V-XX the BHP density in the first wall is listed and compared to the same measure of BHP of natural uranium. The ratios here are not the same as those for the total blanket because of neutron spectral differences. The interesting point here is that the air dilutent required to reduce the activities after one day of decay to MPC levels, is only a factor of 100 to 1000 greater than the dilutent required for the dispersal of 1 cm^3 of natural uranium (a pyrophoric metal which can quite easily ignite during machining or in a high-temperature oxidizing environment).

Turning to the long-term problem of waste storage, we now consider the BHP (water) for CTR structural materials at 100 years or longer. These values range from $<<10^{-25}$ km^3 of water per kW(th) for the V-20Ti alloys, to approximately 10^{-4} km^3/kW(th) for the TZM alloy. The significance of these figures can be more easily understood if they are normalized to 1 cm^3 of blanket material and then compared to the respective figures for uranium metal. On this basis (see Table V-XX), we find that after 100 years of decay the activity in the V-20Ti alloy has dropped to essentially insignificant levels and the activities of Al 2024 and Nb-1Zr have decayed to values comparable to that of natural uranium. The SS 316 and TZM alloys still remain at levels of 10 to 30 above natural uranium. After 10,000 years of decay, the Al and Nb alloy values have not changed significantly, while the SS 316 activity has dropped by a factor of about 100, and the TZM value has only dropped by a factor of two.

The BHP concept only considers the effects of radioisotopes in one individual over his or her lifetime. Since some of the isotopes in both fusion and fission can be around for thousands or millions of years, it is possible that they may enter the food chain several times.

One way to account for such recurrences is to integrate the BHP over the "effective" lifetime of the isotope in question. This can be stated mathematically as an integrated BHP (IBHP) of the ith isotope, defined as:

$$IBHP^{(i)} = (BHP)^{(i)}_{t=0} \cdot \int_0^\infty \exp(-\lambda t)\,dt \quad ,$$

where

$$\lambda = \frac{0.693}{t^i_{1/2}} \quad .$$

Table V-XVIII: Summary of Biological Hazard Potentials (BHP of Air) of Various CTR Structural Materials

Isotope	MPC_{air}[a] (Ci/km^3)	Biological Hazard Potential[b,c] (km^3 of air/kW(th)) SS 316	TZM	V-20Ti	Al 2024	Nb-1Zr
Na24	5				51.2	
Mg27	30	$1.2 \cdot 10^{-3}$			7	
Al26	0.1				$9 \cdot 10^{-4}$	
Al28	30	0.44			3.9	
Ca45	1.0	0.01	$2.6 \cdot 10^{-1}$	4.8		
Sc46	0.8	0.01	$5.9 \cdot 10^{-1}$	10.8		
Sc47	20	$5 \cdot 10^{-4}$	$2.4 \cdot 10^{-2}$	0.44		
Sc48	5	$4 \cdot 10^{-3}$	$2 \cdot 10^{-1}$	7.68		
Sc49	1300		$4 \cdot 10^{-5}$	$7 \cdot 10^{-4}$		
Ti45	340		$6 \cdot 10^{-5}$	$1.3 \cdot 10^{-3}$		
Ti51	850	$2 \cdot 10^{-4}$		0.09		
V 49	250	$4.7 \cdot 10^{-3}$		$6.8 \cdot 10^{-3}$	$8.0 \cdot 10^{-6}$	
V 52	350	0.10		3.0	$1.1 \cdot 10^{-3}$	
Cr51	80	1.3				
Mn53	0.1	$5 \cdot 10^{-5}$			$1.8 \cdot 10^{-7}$	
Mn54	1	53.5			4.4	
Mn56	20	17.6			1.3	
Mn57	30	0.08				
Fe55	30	6.5			$2.1 \cdot 10^{-2}$	
Fe59	2	0.07				
Co57	6	3.2				
Co58	2	45.1				
Co60m	30	0.35				
Co60	0.3	16			1.0	
Ni57	1	3.54				
Ni59	20	$1.5 \cdot 10^{-6}$				
Ni63	0.1	0.36				
Cu64	40				3.7	
Zn65	2				$6 \cdot 10^{-3}$	
Sr89	0.3	$7 \cdot 10^{-3}$				$6 \cdot 10^{-2}$
Sr90	0.03	$1 \cdot 10^{-4}$	$1.2 \cdot 10^{-4}$			$1.2 \cdot 10^{-3}$
Y 90	3		$6.7 \cdot 10^{-3}$			3.9
Y 91	1					0.06
Zr89	0.1	1.8	89.7			27.4
Zr95	1.0	0.04	2.2			
Nb92m[d]	370	$1.1 \cdot 10^{-3}$	$5.7 \cdot 10^{-2}$			2.0
Nb94m[e]	200,000					$2 \cdot 10^{-2}$
Nb94	2		$2.1 \cdot 10^{-7}$			$3 \cdot 10^{-4}$
Nb95m	300	$2.3 \cdot 10^{-4}$	$1.2 \cdot 10^{-2}$			$5.6 \cdot 10^{-2}$
Nb95	3	$7 \cdot 10^{-2}$	2.3			5.5

Table V-XVIII: Summary of Biological Hazard Potentials (BHP of Air) of Various CTR Structural Materials (cont'd)

Isotope	MPC_{air}[a] (Ci/km^3)	Biological Hazard Potential[b,c] (km^3 of air/kW(th)) SS 316	TZM	V-20Ti	Al 2024	Nb-1Zr
Nb96	0.1	0.9	45.9			
Nb97	200	$3.3 \cdot 10^{-4}$	$1.1 \cdot 10^{-2}$			
Mo91	30	$4.2 \cdot 10^{-2}$	2.2			
Mo93	370[f]	$1.4 \cdot 10^{-5}$	$7 \cdot 10^{-4}$			
Mo99	7	4.1	215			
Mo101	30	0.28	14.6			
Tc99m	500	$6 \cdot 10^{-2}$	3.0			
Tc99	2	$1 \cdot 10^{-4}$	$5 \cdot 10^{-3}$			
Tc101	30	0.3	15.0			
Total		156	391	27	73	39

[a] MPC(air): the lower of the soluble or insoluble values is used;

[b] See Table V-XV for corresponding alloy compositions and radioactivity levels;

[c] At shut-down after two years of operation;

[d] 50% isomeric ratio to Nb92m;

[e] 50% isomeric ratio to Nb94m;

[f] Since Mo93 has not been evaluated as to a specific MPC value, we have chosen the value of Nb92m which has a similar half-life and chemistry (but the X-ray is 30 times higher than for Mo93). The low energy decay of Mo93 (30 keV) may, in fact, allow a higher MPC to be assigned, thus alleviating the long-term BHP problem of Mo containing alloys.

Table V-XIX: Summary of Biological Hazard Potentials (BHP of Water) of Various Structural Materials

Isotope	MPC_{water} [a,b] (Ci/km^3)	Biological Hazard Potential [c,d] (km^3 of water/kW(th)) SS 316	TZM	V-20Ti	Al 2024	Nb-1Zr
Na24	30,000				$8.53 \cdot 10^{-3}$	
Mg27						
Al26	3,000				$3.0 \cdot 10^{-8}$	
Al28						
Ca45	9,000	$1.1 \cdot 10^{-6}$	$2.9 \cdot 10^{-5}$	$5.3 \cdot 10^{-4}$		
Sc46	40,000	$2.0 \cdot 10^{-7}$	$1.2 \cdot 10^{-5}$	$2.2 \cdot 10^{-4}$		
Sc47	90,000	$1.1 \cdot 10^{-7}$	$5.3 \cdot 10^{-6}$	$9.8 \cdot 10^{-5}$		
Sc48	30,000	$6.7 \cdot 10^{-7}$	$3.3 \cdot 10^{-5}$	$1.3 \cdot 10^{-3}$		
Sc49						
Ti45	3,000		$6.7 \cdot 10^{-6}$	$1.4 \cdot 10^{-3}$		
Ti51						
V 49	3,000	$4.0 \cdot 10^{-4}$		$5.63 \cdot 10^{-4}$	$6.7 \cdot 10^{-7}$	
V 52						
Cr51	2,000,000	$5.0 \cdot 10^{-5}$				
Mn53	3,000	$1.7 \cdot 10^{-9}$			$6.0 \cdot 10^{-12}$	
Mn54	100,000	$5.4 \cdot 10^{-4}$			$4.4 \cdot 10^{-5}$	
Mn56	100,000	$3.5 \cdot 10^{-3}$			$2.6 \cdot 10^{-4}$	
Mn57	3,000	$8.1 \cdot 10^{-4}$			$2.7 \cdot 10^{-4}$	
Fe55	800,000	$2.4 \cdot 10^{-4}$			$7.9 \cdot 10^{-7}$	
Fe59	50,000	$2.8 \cdot 10^{-6}$				
Co57	400,000	$4.9 \cdot 10^{-5}$				
Co58	90,000	$1.0 \cdot 10^{-3}$				
Co60m						
Co60	30,000	$1.6 \cdot 10^{-4}$			$1.0 \cdot 10^{-5}$	
Ni57	3,000	$1.2 \cdot 10^{-3}$				
Ni59	200,000	$1.5 \cdot 10^{-10}$				
Ni63	30,000	$1.2 \cdot 10^{-6}$				
Cu64	200,000				$7.0 \cdot 10^{-4}$	
Zn65	100,000				$1.2 \cdot 10^{-7}$	
Sr89	3,000	$7.0 \cdot 10^{-7}$				$7.0 \cdot 10^{-6}$
Sr90	300	$1.3 \cdot 10^{-8}$	$1.22 \cdot 10^{-8}$			$1.2 \cdot 10^{-7}$
Y 90	20,000		$1 \cdot 10^{-6}$			$5.9 \cdot 10^{-4}$
Y 91	30,000					$2.0 \cdot 10^{-6}$
Zr89	3,000		$3.0 \cdot 10^{-3}$			$9.1 \cdot 10^{-4}$
Zr95	60,000 [f]	$6.7 \cdot 10^{-7}$	$3.7 \cdot 10^{-5}$			
Nb92m [e]	3,000 [f]	$1.4 \cdot 10^{-4}$	$7 \cdot 10^{-3}$			0.25
Nb94m						
Nb94	3,000	$3.0 \cdot 10^{-12}$	$1.4 \cdot 10^{-10}$			$6.0 \cdot 10^{-8}$
Nb95m	3,000		$1.2 \cdot 10^{-3}$			$5.6 \cdot 10^{-4}$
Nb95	100,000	$2.1 \cdot 10^{-6}$	$6.9 \cdot 10^{-5}$			$1.7 \cdot 10^{-4}$

Table V-XIX: Summary of Biological Hazard Potentials (BHP of Water) of Various Structural Materials (cont'd)

Isotope	MPC_{water}[a,b] (Ci/km^3)	Biological Hazard Potential[c,d] (km^3 of water/kW(th)) SS 316	TZM	V-20Ti	Al 2024	Nb-1Zr
Nb96	3,000		$1.53 \cdot 10^{-3}$			
Nb97	900,000	$7.2 \cdot 10^{-8}$	$2.5 \cdot 10^{-6}$			
Mo91						
Mo93	3,700,000	$1.4 \cdot 10^{-9}$	$7 \cdot 10^{-8}$			
Mo99	40,000	$7.1 \cdot 10^{-4}$	$3.8 \cdot 10^{-2}$			
Mo101						
Tc99m	3,000,000	$1 \cdot 10^{-5}$	$5 \cdot 10^{-4}$			
Tc99	200,000	$1 \cdot 10^{-3}$	$5 \cdot 10^{-8}$			
Tc101						
Total		$1.0 \cdot 10^{-2}$	$5.2 \cdot 10^{-2}$	$4.1 \cdot 10^{-3}$	$1.0 \cdot 10^{-2}$	$2.6 \cdot 10^{-1}$

[a] MPC (water) with insoluble element for general public;

[b] No MPC (water) is given in 10 CFR 20 for isotopes with half-lives less than two hours; all isotopes with no entry for MPC belong to this class;

[c] See Table V-XV for corresponding alloy compositions and radioactivity levels;

[d] At shut-down after two years of operation;

[e] 50 per cent isomeric ratio to Nb92m.

[f] Since there is no evaluation for Nb92m for water, the MPC value for Mo93 is based on the relationship between air and water MPC values for Tc99, which has a similar chemistry, decay sequence, and an even longer half-life.

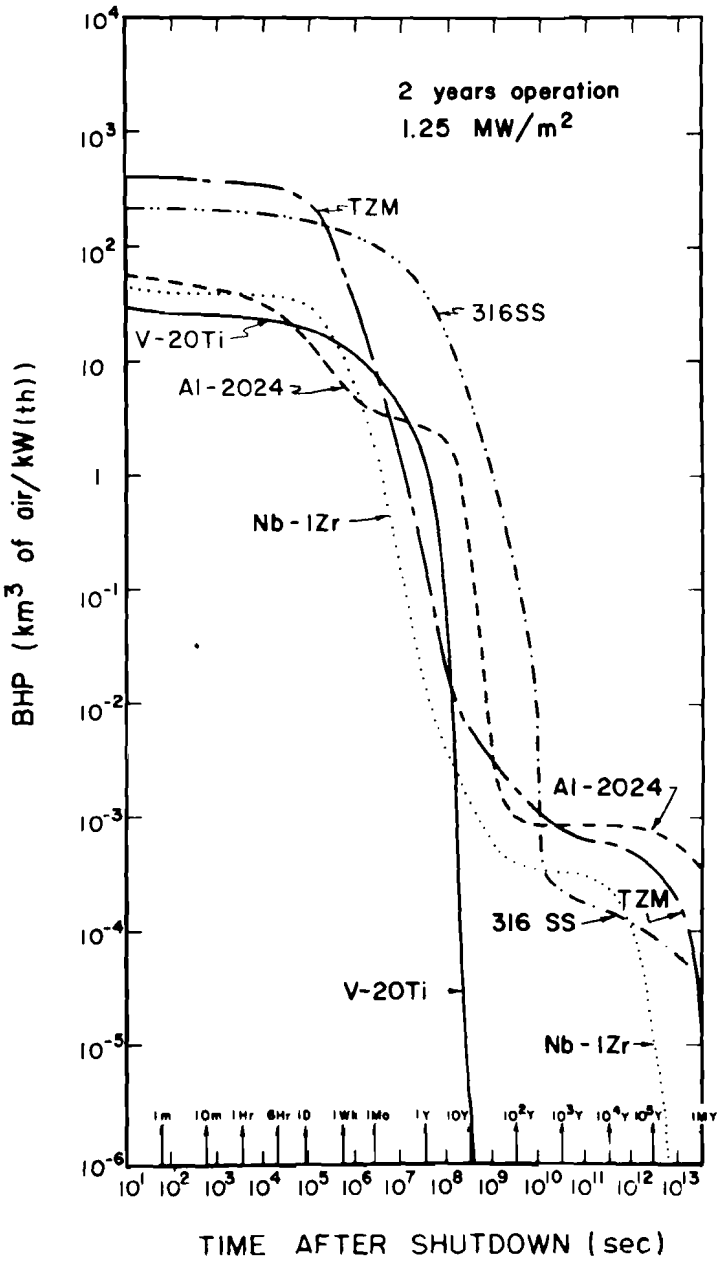

Figure V-16: Biological Hazard Potential (BHP of Air) of Various CTR Structural Materials After Shut-down

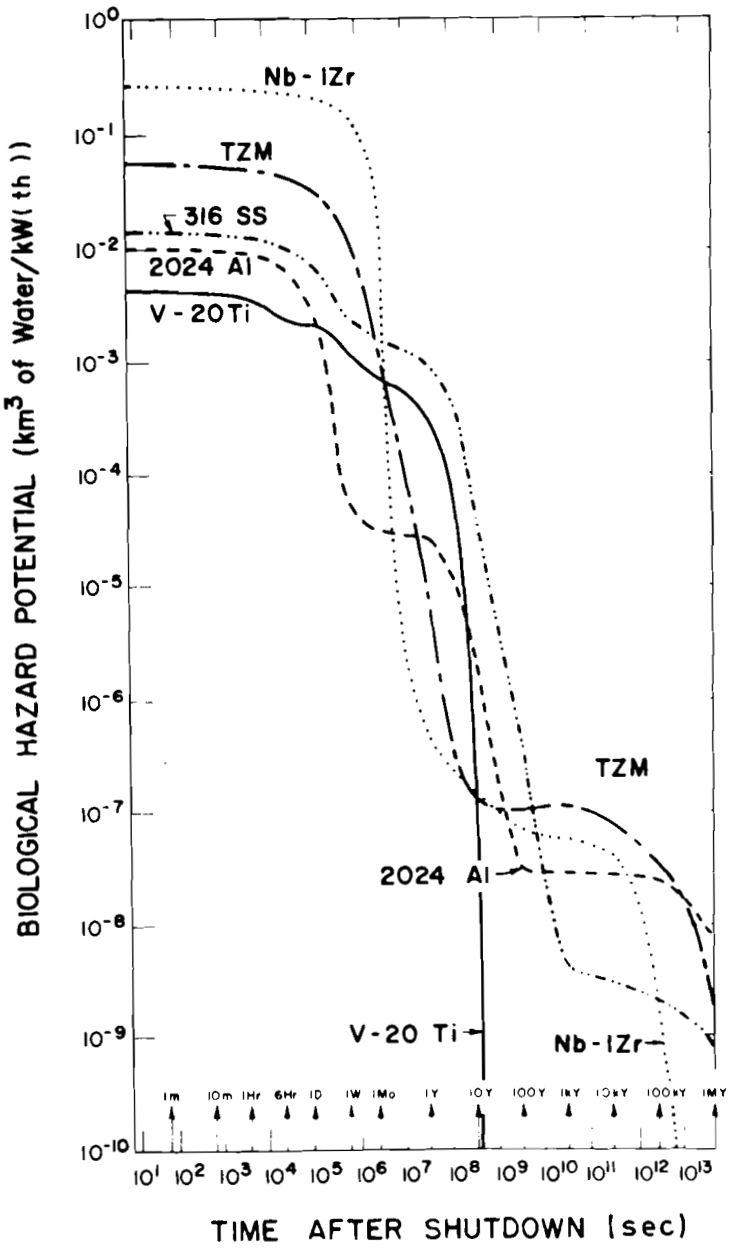

Figure V-17: Biological Hazard Potential (BHP of Water) of Various CTR Structural Materials After Shut-down

Table V-XX: Specific Biological Hazard Potentials of Various CTR Structural Materials

	BHP_{air} (km^3 of air/ cm^3 of material) t = 0	BHP_{water} (km^3 of water/ cm^3 of material) 100 yr	10,000 yr
SS 316	2.0	5.4	0.03
TZM	2.9	1.0	0.7
V-20Ti	0.2	$2 \cdot 10^{-30}$	0
Al 2024	0.55	0.2	0.2
Nb-1Zr	0.36	0.6	0.3
Natural Uranium	$1.3 \cdot 10^{-3}$	0.2	0.2

This definition then gives

$$IBHP^{(i)} = (BHP)^{(i)}_{t=0} \cdot \frac{t^i_{1/2}}{0.693} \quad .$$

It is instructive to note that the MPC values are inversely proportional to the effective half-life $\left(t^e_{1/2}\right)$, which is related to the biological half-life $\left(t^b_{1/2}\right)$ and physical half-life $\left(t^p_{1/2}\right)$ in the following manner:

$$\frac{1}{t^e_{1/2}} = \frac{1}{t^b_{1/2}} + \frac{1}{t^p_{1/2}}$$

Since we are concerned with the isotopes with long half-lives $\left(t^p_{1/2} >> t^b_{1/2}\right)$, the MPC value is inversely proportional to $t^b_{1/2}$. This in turn makes the IBHP value proportional to

$t^{b}_{1/2}\ t^{pi}_{1/2}$. Such a parameter gives an indication of the total burden to society over the entire decay lifetime of a radioisotope and reflects the biological retention in a given generation.

Because of the long time implications of the IBHP index, it makes sense to only consider the situation with respect to waste disposal and hence the IBHP for water. Individual and collective values for the various CTR isotopes are given in Table V-XXI and V-XXII. For the case of water dilution, Mo93 dominates both the TZM alloy and the SS 316. The Ca45 and V49 isotopes account for most of the IBHP values in the V-Ti alloys, while Al26 accounts for more than 99 per cent of the hazard index in the Al alloy. (Note, it is not the alloying elements which dominate the Al alloys but rather the base metal itself). Finally, the Nb92m isotope is the major factor in the Nb-1Zr alloy with regard to the water IBHP value.

The IBHP values for water are summarized in Table V-XXIII, normalized to a cm^3 of first-wall structure. The appropriate values for a cm^3 of uranium metal are included for comparison. Such a comparison is not entirely free from misinterpretation because, obviously, there is no correlation between the number of cm^3 of first-wall material and the amount of metallic uranium in the world. However, it does lend some perspective to the *total* societal burden associated with the generation of radioactive waste in a fusion reactor. These calculations show that over the entire time required for the radioactivity to decay away, irradiated fusion-reactor structural materials vary by only a factor of 30 in the value of IBHP for water, where Al has the highest index, followed by TZM, SS 316, Nb-1Zr, and V-20Ti in that order. The IBHP values are one to two orders of magnitude higher than those of natural-uranium metal.

In summary, it can be said that practically all fusion-reactor structural designs will have radioactivity levels of 1000 Ci/kW(th) after modest irradiation times (one month). The decay of the radioisotopes varies from alloy to alloy but, except for systems which contain Mo, the specific activity (Ci/cm^3 of metal) after 100 years of decay is no worse, and in some cases lower, than the activity associated with one cm^3 of uranium metal. As regards the biological hazard potential, it is obvious that high values do exist immediately after plant shut-down, but again the hazard index associated with one cm^3 of irradiated V-20Ti, Al 2024, and Nb-1Zr after 100 years of decay is equal to or less than that associated with a cm^3 of natural-uranium metal. Steels have a BHP (water) which is 25 times that of natural U, and TZM is a factor of five higher than U one hundred years after shut-down. Finally, the IBHP of the irradiated first-wall materials is only one to two orders of magnitude higher than the IBHP of the same volume of natural-uranium metal.

Table V-XXI: Integrated BHP (Air) Index for Potential Fusion Blanket Structural Materials (km^3_{air} · sec/kW(th))

Isotope	$\frac{t_{1/2}}{0.693}$ sec	SS 316	TZM	V-20Ti	Al 2024	Nb-1Zr
Na24	$7.8 \cdot 10^{4}$				$4.0 \cdot 10^{6}$	
Mg27	$8.2 \cdot 10^{2}$	$9.8 \cdot 10^{-1}$			$5.7 \cdot 10^{3}$	
Al26	$3.4 \cdot 10^{13}$				$3.1 \cdot 10^{10}$	
Al28	$2.0 \cdot 10^{2}$	$8.8 \cdot 10^{1}$			$7.8 \cdot 10^{2}$	
Ca45	$2.1 \cdot 10^{7}$	$2.1 \cdot 10^{5}$	$5.5 \cdot 10^{5}$	$1.0 \cdot 10^{8}$		
Sc46	$1.0 \cdot 10^{7}$	$1.0 \cdot 10^{5}$	$5.9 \cdot 10^{6}$	$1.1 \cdot 10^{8}$		
Sc47	$4.3 \cdot 10^{5}$	$2.2 \cdot 10^{2}$	$1.0 \cdot 10^{4}$	$1.9 \cdot 10^{5}$		
Sc48	$2.3 \cdot 10^{5}$	$9.2 \cdot 10^{2}$	$4.6 \cdot 10^{4}$	$1.8 \cdot 10^{6}$		
Sc49	$5.0 \cdot 10^{3}$		$2.0 \cdot 10^{-1}$	3.5		
Ti45	$1.6 \cdot 10^{4}$		$9.6 \cdot 10^{-1}$	$2.1 \cdot 10^{1}$		
Ti51	$5.0 \cdot 10^{2}$	$1.0 \cdot 10^{-1}$		$4.5 \cdot 10^{1}$		
V 49	$4.1 \cdot 10^{7}$	$1.9 \cdot 10^{5}$		$2.8 \cdot 10^{5}$	$3.3 \cdot 10^{2}$	
V 52	$3.2 \cdot 10^{2}$	$3.2 \cdot 10^{1}$		$9.6 \cdot 10^{2}$	$3.5 \cdot 10^{-1}$	
Cr51	$3.5 \cdot 10^{6}$	$4.6 \cdot 10^{6}$				
Mn53	$8.6 \cdot 10^{13}$	$4.3 \cdot 10^{9}$			$1.5 \cdot 10^{7}$	
Mn54	$3.8 \cdot 10^{8}$	$2.0 \cdot 10^{10}$			$1.7 \cdot 10^{9}$	
Mn56	$1.3 \cdot 10^{4}$	$2.3 \cdot 10^{5}$			$1.7 \cdot 10^{4}$	
Mn57	$8.7 \cdot 10^{5}$	$7.0 \cdot 10^{4}$				
Fe55	$1.2 \cdot 10^{8}$	$7.8 \cdot 10^{8}$			$2.5 \cdot 10^{6}$	
Fe59	$5.7 \cdot 10^{6}$	$4.0 \cdot 10^{5}$				
Co57	$3.4 \cdot 10^{7}$	$1.1 \cdot 10^{8}$				
Co58	$8.9 \cdot 10^{6}$	$4.0 \cdot 10^{8}$				
Co60m	$9.1 \cdot 10^{2}$	$3.2 \cdot 10^{2}$				
Co60	$2.4 \cdot 10^{8}$	$3.9 \cdot 10^{9}$				
Ni57	$1.9 \cdot 10^{5}$	$6.7 \cdot 10^{5}$				
Ni59	$3.6 \cdot 10^{12}$	$5.4 \cdot 10^{6}$				
Ni63	$4.2 \cdot 10^{9}$	$1.5 \cdot 10^{9}$				
Cu64	$6.6 \cdot 10^{4}$				$2.4 \cdot 10^{5}$	
Zn65	$3.1 \cdot 10^{7}$				$1.9 \cdot 10^{5}$	
Sr89	$6.6 \cdot 10^{6}$	$4.6 \cdot 10^{4}$				
Sr90	$1.3 \cdot 10^{9}$	$1.3 \cdot 10^{5}$	$1.6 \cdot 10^{5}$			$4.0 \cdot 10^{5}$
Y 90	$3.3 \cdot 10^{5}$		$2.2 \cdot 10^{3}$			$1.6 \cdot 10^{6}$
Y 91	$7.3 \cdot 10^{6}$					$1.3 \cdot 10^{6}$
Zr89	$4.1 \cdot 10^{5}$	$7.4 \cdot 10^{5}$	$3.7 \cdot 10^{7}$			$4.4 \cdot 10^{5}$
Zr95	$8.2 \cdot 10^{6}$	$3.3 \cdot 10^{5}$	$1.8 \cdot 10^{7}$			$1.1 \cdot 10^{7}$
Nb92m	$1.3 \cdot 10^{6}$	$1.4 \cdot 10^{3}$	$7.4 \cdot 10^{4}$			$2.6 \cdot 10^{6}$
Nb94m	$5.5 \cdot 10^{2}$					$1.1 \cdot 10^{1}$
Nb94	$9.1 \cdot 10^{11}$		$1.9 \cdot 10^{5}$			$2.7 \cdot 10^{8}$
Nb95m	$4.7 \cdot 10^{5}$	$1.1 \cdot 10^{2}$	$5.6 \cdot 10^{3}$			$2.6 \cdot 10^{4}$
Nb95	$4.4 \cdot 10^{6}$	$3.1 \cdot 10^{5}$	$1.0 \cdot 10^{7}$			$2.4 \cdot 10^{7}$

Table V-XXI: Integrated BHP (Air) Index for Potential Fusion Blanket Structural Materials (Cont'd) (km^3_{air} · sec/kW(th))

Isotope	$\frac{t_{1/2}}{0.693}$ sec	SS 316	TZM	V-20Ti	Al 2024	Nb-1Zr
Nb96	$1.2 \cdot 10^{5}$	$1.1 \cdot 10^{5}$	$5.5 \cdot 10^{6}$			
Nb97	$6.2 \cdot 10^{3}$	2.0	$6.8 \cdot 10^{1}$			
Mo91	$1.3 \cdot 10^{3}$	$5.5 \cdot 10^{1}$	$2.9 \cdot 10^{3}$			
Mo93	$4.5 \cdot 10^{11}$	$6.0 \cdot 10^{6}$	$3.0 \cdot 10^{8}$			
Mo99	$3.5 \cdot 10^{5}$	$1.4 \cdot 10^{6}$	$7.5 \cdot 10^{7}$			
Mo101	$1.3 \cdot 10^{3}$	$3.6 \cdot 10^{2}$	$1.9 \cdot 10^{4}$			
Tc99m	$3.6 \cdot 10^{4}$	$2.2 \cdot 10^{3}$	$1.1 \cdot 10^{5}$			
Tc99	$9.8 \cdot 10^{12}$	$9.5 \cdot 10^{8}$	$4.9 \cdot 10^{10}$			
Tc101	$1.2 \cdot 10^{3}$	$3.6 \cdot 10^{2}$	$1.8 \cdot 10^{4}$			
Total		$3.1 \cdot 10^{10}$	$4.9 \cdot 10^{10}$	$2.1 \cdot 10^{8}$	$3.3 \cdot 10^{10}$	$3.1 \cdot 10^{8}$

Table V-XXII: Integrated BHP (Water) Index for Potential Fusion Blanket Structural Materials (km^3_{water} · sec/kW(th))

Isotope	$\frac{t_{1/2}}{0.693}$ sec	SS 316	TZM	V-20Ti	Al 2024	Nb-1Zr
Na24	$7.8 \cdot 10^{4}$				$6.7 \cdot 10^{2}$	
Mg27	$8.2 \cdot 10^{2}$					
Al26	$3.4 \cdot 10^{13}$				$1.0 \cdot 10^{6}$	
Al28	$2.0 \cdot 10^{2}$					
Ca45	$2.1 \cdot 10^{7}$	$2.3 \cdot 10^{1}$	$6.1 \cdot 10^{2}$	$1.1 \cdot 10^{4}$		
Sc46	$1.0 \cdot 10^{7}$	2.0	$1.2 \cdot 10^{2}$	$2.2 \cdot 10^{3}$		
Sc47	$4.3 \cdot 10^{5}$	$4.7 \cdot 10^{-2}$	2.3	$4.2 \cdot 10^{1}$		
Sc48	$2.3 \cdot 10^{5}$	$1.5 \cdot 10^{-1}$	7.6	$3.0 \cdot 10^{2}$		
Sc49	$5.0 \cdot 10^{3}$					
Ti45	$1.6 \cdot 10^{4}$		$1.1 \cdot 10^{-1}$	$2.2 \cdot 10^{1}$		

Table V-XXII: Integrated BHP (Water) Index for Potential Fusion Blanket Structural Materials (Cont'd)
(km^3_{water} · sec/kW(th))

Isotope	$\frac{t_{1/2}}{0.693}$ sec	SS 316	TZM	V-20Ti	Al 2024	Nb-1Zr
Ti51	$5.0 \cdot 10^{2}$					
V 49	$4.1 \cdot 10^{7}$	$1.6 \cdot 10^{4}$		$2.3 \cdot 10^{4}$	$2.7 \cdot 10^{1}$	
V 52	$3.2 \cdot 10^{2}$					
Cr51	$3.5 \cdot 10^{6}$	$1.8 \cdot 10^{2}$				
Mn53	$8.6 \cdot 10^{13}$	$1.5 \cdot 10^{5}$			$5.2 \cdot 10^{2}$	
Mn54	$3.8 \cdot 10^{8}$	$2.1 \cdot 10^{5}$			$1.7 \cdot 10^{4}$	
Mn56	$1.3 \cdot 10^{4}$	$4.6 \cdot 10^{1}$			3.4	
Mn57	$8.7 \cdot 10^{5}$	$7.0 \cdot 10^{2}$				
Fe55	$1.2 \cdot 10^{8}$	$2.9 \cdot 10^{4}$			$9.5 \cdot 10^{1}$	
Fe59	$5.7 \cdot 10^{6}$	$1.6 \cdot 10^{1}$				
Co57	$3.4 \cdot 10^{7}$	$1.7 \cdot 10^{3}$				
Co58	$8.9 \cdot 10^{6}$	$8.9 \cdot 10^{3}$				
Co60m	$9.1 \cdot 10^{2}$					
Co60	$2.4 \cdot 10^{8}$	$3.8 \cdot 10^{4}$			$2.4 \cdot 10^{3}$	
Ni57	$1.9 \cdot 10^{5}$	$2.3 \cdot 10^{2}$				
Ni59	$3.6 \cdot 10^{12}$	$5.4 \cdot 10^{2}$				
Ni63	$4.2 \cdot 10^{9}$	$5.0 \cdot 10^{3}$				
Cu64	$6.6 \cdot 10^{4}$				$4.6 \cdot 10^{1}$	
Zn65	$3.1 \cdot 10^{7}$				3.7	
Sr89	$6.6 \cdot 10^{6}$	4.6				$4.6 \cdot 10^{1}$
Sr90	$1.3 \cdot 10^{9}$	$1.7 \cdot 10^{1}$	$1.6 \cdot 10^{1}$			$1.6 \cdot 10^{2}$
Y 90	$3.3 \cdot 10^{5}$		$3.3 \cdot 10^{-1}$			$1.9 \cdot 10^{2}$
Y 91	$7.3 \cdot 10^{6}$					$1.5 \cdot 10^{1}$
Zr89	$4.1 \cdot 10^{5}$		$1.2 \cdot 10^{3}$			$3.7 \cdot 10^{2}$
Zr95	$8.2 \cdot 10^{6}$	5.5	$3.0 \cdot 10^{2}$			
Nb92m	$1.3 \cdot 10^{6}$	$1.8 \cdot 10^{2}$	$9.1 \cdot 10^{3}$			$3.3 \cdot 10^{5}$
Nb94m	$5.5 \cdot 10^{2}$					
Nb94	$9.1 \cdot 10^{11}$	2.7	$1.3 \cdot 10^{2}$			$5.5 \cdot 10^{4}$
Nb95m	$4.7 \cdot 10^{5}$		$5.6 \cdot 10^{2}$			$2.6 \cdot 10^{3}$
Nb95	$4.4 \cdot 10^{6}$	9.2	$3.0 \cdot 10^{2}$			$7.5 \cdot 10^{2}$
Nb96	$1.2 \cdot 10^{5}$	3.6	$1.8 \cdot 10^{2}$			
Nb97	$6.2 \cdot 10^{3}$	$4.5 \cdot 10^{4}$	$1.6 \cdot 10^{-2}$			
Mo91	$1.3 \cdot 10^{3}$					
Mo93	$4.5 \cdot 10^{11}$	$6.0 \cdot 10^{2}$	$3.0 \cdot 10^{4}$			
Mo99	$3.5 \cdot 10^{5}$	$2.5 \cdot 10^{2}$	$1.3 \cdot 10^{4}$			
Mo101	$1.3 \cdot 10^{3}$					
Tc99m	$3.6 \cdot 10^{4}$	$3.6 \cdot 10^{-1}$	$1.8 \cdot 10^{1}$			
Tc99	$9.8 \cdot 10^{12}$	$1.0 \cdot 10^{4}$	$5.0 \cdot 10^{5}$			
Tc101	$1.2 \cdot 10^{3}$					
Total		$5.1 \cdot 10^{5}$	$5.4 \cdot 10^{5}$	$3.68 \cdot 10^{4}$	$1.04 \cdot 10^{6}$	$3.87 \cdot 10^{5}$

Table V-XXIII: Summary of IBHP (Water) Values of Various CTR Structural Materials
(km^3 of water/cm^3 of material)

	IBHP (Water)
SS 316	$23 \cdot 10^4$
TZM	$24 \cdot 10^4$
V-20Ti	$1.6 \cdot 10^4$
Al 2024	$46 \cdot 10^4$
Nb-1Zr	$17 \cdot 10^4$
Natural Uranium	$2. \cdot 10^3$

3.3 Radioactivity in Fusion Reactor Coolants

Thus far, only two primary coolants have been seriously considered for D-T fusion reactors:

(a) helium gas;

(b) liquid lithium.

The only significant activity induced in pure helium is the He4 (n,γ) He5 reaction. Since He5 decays with a half-life of 10^{-21} sec, the activity will build up to approximately 0.5 Ci/W at normal temperatures and pressures of a helium coolant, because of the short half-life.

Pure lithium generates no significant radioisotopes other than the tritium, which has already been treated in an earlier section. However, as with any material, there are normally a large variety of impurity atoms which can become radioactive. P.J. Persiani [V-33] has investigated this problem and notes that lithium normally contains the impurities listed in Table V-XXIV. The activation of such impurities could lead to problems, especially for elements such as Fe, Ni, Cr, and Ta. The resulting radiation levels could even be magnified if these impurities tended to concentrate in specific parts of the reactor such as valves, pumps, or heat exchangers.

An idea of the radioactivity induced in a typical reactor like UWMAK-I [V-14] can be appreciated by noting that there are $2 \cdot 10^6$ kg of coolant in that system. This amount of coolant contains approximately 200 kg each of Ca, Na, K, Fe, Ni, Cr, and Ta. It may also contain $6 \cdot 10^3$ kg of F and Cl. Since

Table V-XXIV: Tpyical Impurities Found in High Purity Lithium

Element	Maximum Concentration (ppm per weight)	Approximate Activity After Two Years of Exposure (Ci/kW(th))
Si	50	low
O	100	low
N	500	low
Ca	100	<1
Na	100	<2
K	100	<2
Fe	100	<1
Ni	100	<1
Cr	100	<1
Ta	100	<0.1
F	3000	<10
Cl	3000	<10
		Total <30

only one-half of this Li is in the reactor at any one time, and 20 per cent of that is in the first 10 cm (highest flux region), one can calculate the *approximate* level of activity after two years of irradiation. These values are also listed in Table V-XXIV, which shows that K, Na, F, and Cl have the highest radioactivity levels. The total induced radioactivity is low compared to that of the structure, and might be discounted by the fact that special purification techniques would be used to remove trace elements which give measurable radioactivity.

One must also consider the activity of corrosion products and sputtered atoms into the coolant. Studies of the stainless steel-lithium system have predicted corrosion rates of 1500 to 2500 kg per year. This radioactive material could add another 1 Ci/kW(th) to the overall radiation levels. Similar effects, although lower in absolute magnitude, will occur for the refractory metals [V-14].

Aside from the radiation levels that may exist outside the reactor, the radioactive impurities and corrosion products represent a severe hazard in the unlikely event of a rupture of

the primary coolant and loop, and the ignition of the molten Li. The oxides of many of these elements (e.g. MoO_2, TaO_2, NbO_2, SrO_2) are quite volatile and could be widely dispersed during a fire. In fact, this potential fire hazard is the most plausible mechanism for the release of radioactive metallic isotopes into the environment.

In summary, the radioactivity in the CTR coolant is only important if liquid Li is used, and then only that which comes from impurities (which could be removed prior to or during operation) or from corrosion (which could be removed by appropriate clean-up techniques). However, an accident in the primary loop could have serious consequences by the release of volatile isotopes.

3.4 Radiation Levels Associated With Shield and Magnet Activation

Most fusion systems are designed to reduce the neutron flux to the shields by at least two orders of magnitude of that incident on the reactor first wall. However, this number of neutrons activating the shield is still large, and even with another 10^3 to 10^4 reduction in the neutron flux to the magnets, the radiation levels outside the reactor can be intolerable. These levels are more of a hindrance to maintenance than a question of safety, but the fact still remains that there will be a large volume of metal with low-activity density which will have to be properly processed and disposed of in a government repository.

It is not fruitful to quote specific numbers at this time for shield and magnet activation, but it should be pointed out that most of the radiation present outside a fusion reactor comes from the last few centimeters of the shield and the magnet structure, and *not* from the blanket materials. The gamma radiation outside the reactor coming from the blanket is almost negligible in a properly designed system. W.F. Vogelsang has recently calculated the radiation levels outside a near-term test reactor (TETR) and the UWMAK-I reactor [V-34]. His results are given in Table V-XXV and they show that, depending on the shield design (see [V-34] for more details), the radiation levels from the shield and the magnet structure can be as high as one to four rem per hour at reactor shut-down, and 0.02 to 1 rem/h after one week of decay. Obviously there is much room for design innovations, but the reader should recall that a great amount of radioisotopes will be generated outside the blanket (see Subsection V.3.2).

Table V-XXV: Calculated Dose Rate Outside Reactor Shields in Typical TOKAMAK Fusion Reactors [V-34]

Time After Shut-down	Dose Rate (mrem/h)	
	TETR (1.1 MW/m^2)	UWMAK-I (1.25 MW/m^2)
0	790	4072
1 hour	530	3068
1 day	22	1165
1 week	18	1084
1 month	16	870
1 year	7	168

4. COMPILATION OF RADIOACTIVITY AND ASSOCIATED INDICES FOR FISSION AND FUSION SYSTEMS

The three methods of calculating induced radioactivity can now be used to compare fission and fusion radioactive inventories. It is most helpful to compile the Ci/kW(th) and BHP values as a function of time after shut-down, and the IBHP values for water. Before, a few qualifying statements must be made:

(1) When comparing long-term inventories, one must take into account that most of the Pu isotopes will be recycled through the fuel cycle. Therefore, the long-term Ci/kW(th) or BHP values for fission must be reduced by this separation factor;

(2) The same reasoning as used in *(1)* applies to fusion systems, where most of the tritium would be removed from a plant after it is shut down and transferred to other reactors.

(3) A comparison between an SS 316 LMFBR and an SS 316 fusion reactor will be made, although the numbers could be scaled to other CTR materials by the use of Tables V-XVI to V-XIX.

A summary of the radioactivity associated with all the fusion concepts and the LMFBR is contained in Table V-XXVI and displayed in Figure V-18. For comparable SS 316 systems we see that at shut-down the activities in the fission reactor are roughly five times higher than those in a fusion reactor. After reprocessing, the activity of SS 316 fusion reactors will be about the same as for the fuel, coolant, and cladding wastes in an LMFBR. This equality actually persists for approximately 10 years, after which the activity from the long-lived fission products and actinide wastes keeps the fission-reactor wastes approximately one order of magnitude higher than the fusion-reactor wastes. As stated in the Introduction, the next step in a line of reasoning that ultimately leads to a realistic comparison can be made on the basis of the BHP values, and these are summarized in Tables V-XXVII and V-XXVIII, and plotted in Figures V-19 and V-20.

At shut-down, the inhalation hazard index (air) for fission reactors is about 30 to 40 times higher than that of a D-T fusion reactor with SS 316 structure. This advantage persists over the entire lifetime of the isotopes, but narrows to only a factor of about six after three months of decay. Thereafter, the inhalation hazard index for fusion becomes a decreasingly smaller fraction of that for fission, so that after approximately 300 years of decay it is five orders of magnitude lower. However, the inhalation hazard index is probably only meaningful for the first year after shut-down (i.e. during an accident, or reprocessing for ultimate disposal).

After the radioisotopes have been prepared for long-term waste storage, the ingestion hazard index becomes important. The appropriate figures for fission and fusion again show that fusion systems have a distinct advantage. Figure V-20 shows that, at the time of reprocessing, the ingestion hazard for fusion reactor components is a factor of five lower than for fission; this advantage increases to a factor of 100 after 300 years of decay, and remains at approximately the same level for several million years.

A final word of caution is added here to stress again that we have not considered *how* these radioisotopes would enter the environment, but only analyzed what would be required *if* they all did.

Finally, we can compare the integrated ingestion-hazard index IBHP (water) for fission and fusion to obtain a rough idea of the total burden these two energy sources represent to society. The values are listed in Table V-XXIX; again, if all the radioisotopes from a SS 316 reactor were released the burden they represent would be less by a factor of ten than if all the radioisotopes in a fission reactor were released. It is interesting to note that the fission IBHP (water) is dominated (at roughly 80 per cent) by I129 in the reprocessing stream. On the fusion side, two isotopes (Mn53 and Mn54) account for 70 per cent of the IBHP (water) for SS 316.

Table V-XXVI: Decay of Induced Radioactivity in Fission and Fusion System After Shut-down (Ci/kW(th))

	t = 0	t = 10^6	t = 10^7	t = 10^8	t = 10^{10}	t = 10^{12}	t = 10^{14} (sec)
Fusion[a]							
T_2	120	120	——— presume T_2 is removed from the plant ———				
SS 316	1062	360	220	60	$2 \cdot 10^{-2}$	$2 \cdot 10^{-4}$	$5 \cdot 10^{-6}$
TZM	4120	320	8	0.8	0.3	$1 \cdot 10^{-1}$	$4 \cdot 10^{-7}$
V-20Ti	1261	5	4	$5 \cdot 10^{-2}$	≈0	-	-
Al 2024	884	8	4	1.4	$9 \cdot 10^{-5}$	$9 \cdot 10^{-5}$	$5 \cdot 10^{-6}$
Nb-1Zr	5155	300	20	$2 \cdot 10^{-3}$	$5 \cdot 10^{-4}$	$2 \cdot 10^{-4}$	≈0
Fission[a]							
SS 316	40	40	9	2	$1 \cdot 10^{-4}$	$1.5 \cdot 10^{-5}$	$1 \cdot 10^{-7}$
Na	44	12	0.4	0.18	≈0	-	-
Actinides[b]	1500	100	50	20	5	$2 \cdot 10^{-3}$	$2 \cdot 10^{-8}$
Fission Products	4000	600	200	20	0.01	$5 \cdot 10^{-4}$	$4 \cdot 10^{-5}$

[a] Basis: two years of operation;

[b] Reprocessing removes 99 per cent of Pu isotopes for reinsertion into the reactor after 200 days.

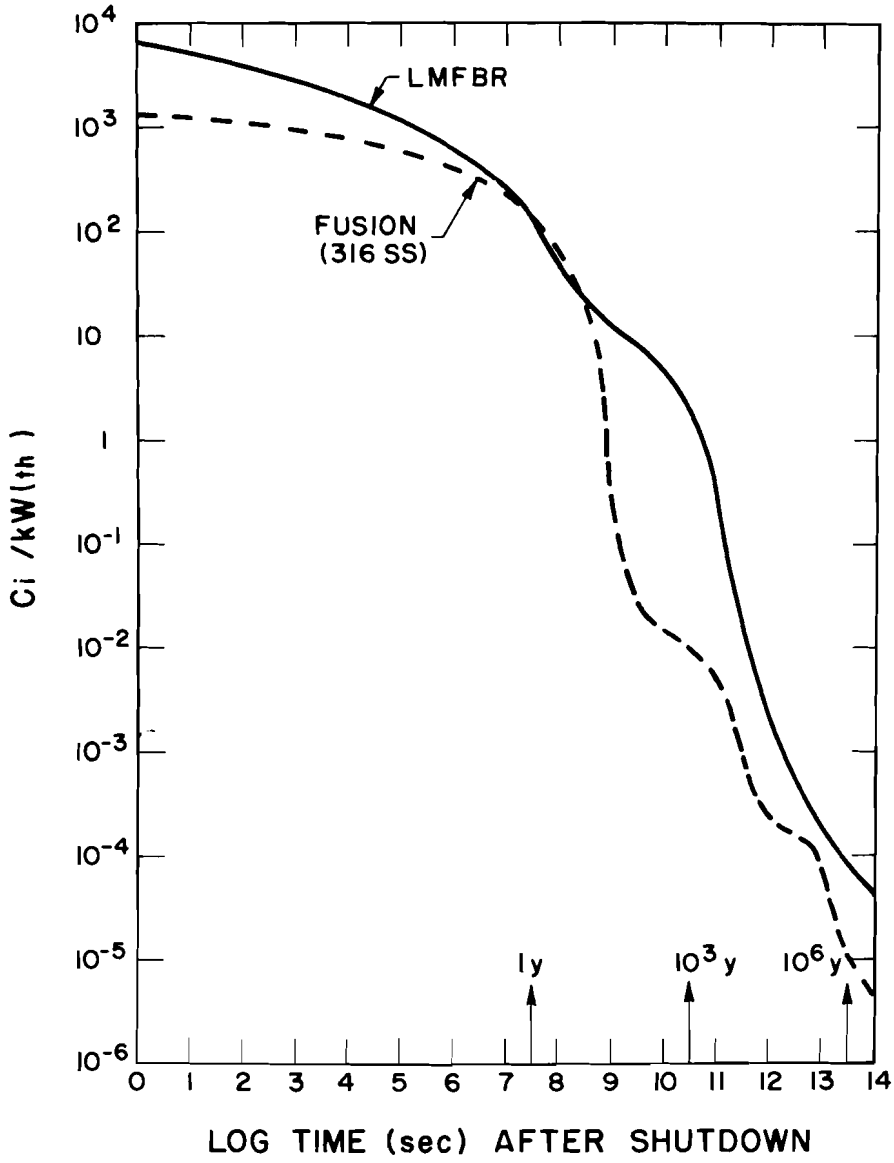

Figure V-18: Comparison of Radioactivity Inventory For Fission and Fusion Reactors With SS 316 Structure

Table V-XXVII: Decay of BHP (Air) for Fission and Fusion Systems (km^3 of air/kW(th))

	t = 0	t = 10^6	t = 10^7	t = 10^8	t = 10^{10}	t = 10^{12}	t = 10^{14} (sec)
Fusion							
T_2	0.6	0.6	——— presume T_2 is removed from the plant ———				
SS 316	156	120	80	20	10^{-3}	$8 \cdot 10^{-5}$	$<1 \cdot 10^{-5}$
TZM	391	30	1	$2.5 \cdot 10^{-2}$	$2 \cdot 10^{-3}$	$5 \cdot 10^{-3}$	$<1 \cdot 10^{-6}$
V-20Ti	27	12	4	$2 \cdot 10^{-3}$	≈ 0	-	-
Al 2024	73	5	3	2	$9 \cdot 10^{-4}$	$9 \cdot 10^{-4}$	$5 \cdot 10^{-5}$
Nb-1Zr	39	7	0.1	$2 \cdot 10^{-3}$	$5 \cdot 10^{-4}$	$1 \cdot 10^{-4}$	$<<1 \cdot 10^{-6}$
Fission							
SS 316	20	20	8	4	$4 \cdot 10^{-5}$	$4 \cdot 10^{-6}$	$<<1 \cdot 10^{-6}$
Na	9.9	1.3	1.3	0.56	≈ 0	-	-
Actinides [a]	5125	5100	400	300	100	4	<0.1
Fission Products	530		140	20	1	$1 \cdot 10^{-4}$	$<1 \cdot 10^{-5}$
Comparison of Similar Systems							
Fusion SS 316	157	121	80	20	$1 \cdot 10^{-3}$	$8 \cdot 10^{-5}$	$1 \cdot 10^{-5}$
Fission SS 316	5685	$\approx$5300	550	325	100	4	0.1

[a] Assume fuel is reprocessed

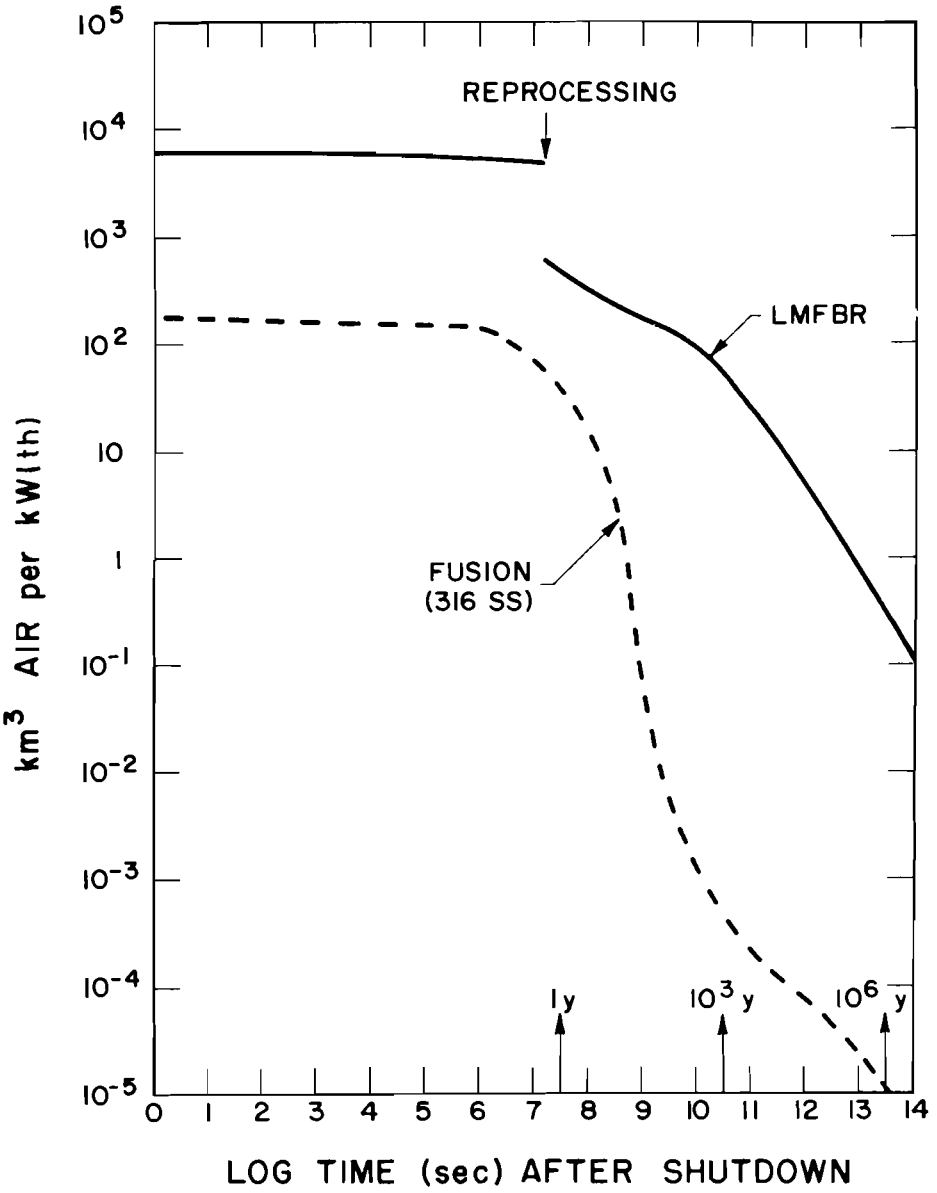

Figure V-19: Comparison of Inhalation Hazard Index of an LMFBR and a D-T Fusion Reactor with SS 316 Structure

Table V-XXVIII: Decay of BHP (Water) For Fission and Fusion System (km^3 of water/kW(th))

	t = 0	t = 10^6	t = 10^7	t = 10^8	t = 10^{10}	t = 10^{12}	t = 10^{14} (sec)
Fusion							
T_2	$7 \cdot 10^{-6}$	$7 \cdot 10^{-6}$	——— presume T_2 is removed from the plant ———				
SS 316	$1 \cdot 10^{-2}$	$2 \cdot 10^{-3}$	$1.3 \cdot 10^{-3}$	$7 \cdot 10^{-4}$	$4 \cdot 10^{-8}$	$3 \cdot 10^{-9}$	$1 \cdot 10^{-9}$
TZM	$5.5 \cdot 10^{-2}$	$1 \cdot 10^{-2}$	$2 \cdot 10^{-4}$	$1 \cdot 10^{-6}$	$3 \cdot 10^{-7}$	$5 \cdot 10^{-8}$	$<<1 \cdot 10^{-10}$
V-20Ti	$4.1 \cdot 10^{-3}$	$1 \cdot 10^{-3}$	$6 \cdot 10^{-4}$	$1 \cdot 10^{-4}$	-	-	-
Al 2024	$1.1 \cdot 10^{-2}$	$5 \cdot 10^{-5}$	$3 \cdot 10^{-5}$	$1.5 \cdot 10^{-5}$	$3 \cdot 10^{-8}$	$3 \cdot 10^{-8}$	$1 \cdot 10^{-8}$
Nb-1Zr	$2.6 \cdot 10^{-1}$	$1.2 \cdot 10^{-1}$	$5 \cdot 10^{-6}$	$3 \cdot 10^{-7}$	$5 \cdot 10^{-8}$	$2 \cdot 10^{-8}$	-
Fission							
SS 316	$2 \cdot 10^{-3}$	$3 \cdot 10^{-4}$	$2 \cdot 10^{-4}$	$4 \cdot 10^{-5}$	$<1 \cdot 10^{-9}$	$<1 \cdot 10^{-10}$	$<<1 \cdot 10^{-10}$
Na	$1.5 \cdot 10^{-3}$	$4 \cdot 10^{-4}$	$1 \cdot 10^{-5}$	$6 \cdot 10^{-6}$	-	-	-
Actinides	$4 \cdot 10^{-3}$		$1 \cdot 10^{-4}$	$1 \cdot 10^{-5}$	$4 \cdot 10^{-6}$	$1 \cdot 10^{-7}$	
Fission Products	$9 \cdot 10^{-2}$	$\approx 1 \cdot 10^{-2}$	$5 \cdot 10^{-3}$	$2 \cdot 10^{-3}$	$4 \cdot 10^{-8}$	$8 \cdot 10^{-9}$	
Comparison of Similar Systems							
Fusion SS 316	$1 \cdot 10^{-2}$	$2 \cdot 10^{-3}$	$1.3 \cdot 10^{-3}$	$7 \cdot 10^{-4}$	$4 \cdot 10^{-8}$	$3 \cdot 10^{-9}$	$1 \cdot 10^{-9}$
Fission SS 316	$1 \cdot 10^{-1}$	$\approx 1 \cdot 10^{-2}$	$5.3 \cdot 10^{-3}$	$2 \cdot 10^{-3}$	$4 \cdot 10^{-6}$	$1 \cdot 10^{-7}$	

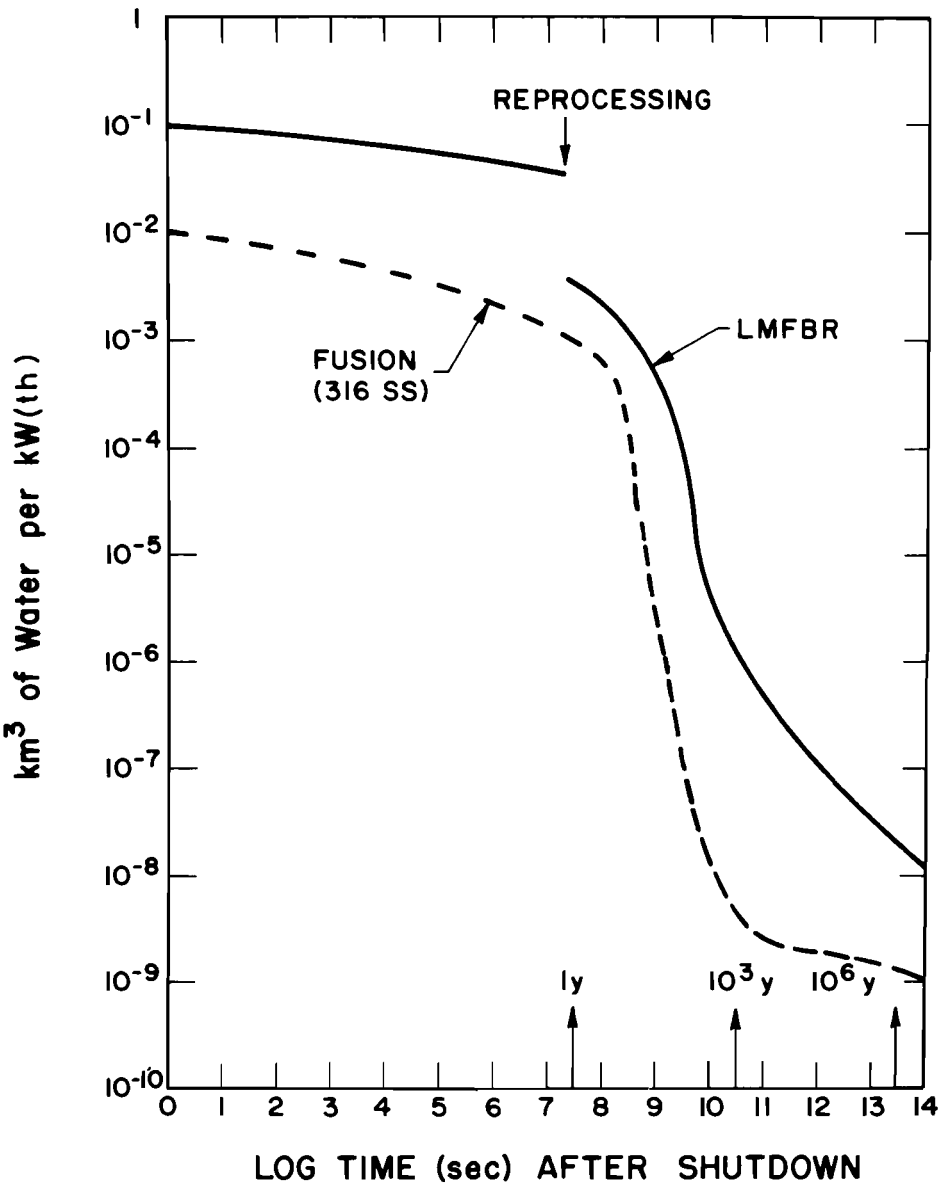

Figure V-20: Comparison of Ingestion Hazard Index of an LMFBR and a D-T Fusion Reactor with SS 316 Structure

Table V-XXIX: Comparison of Integrated BHP of an LMFBR[a] and a D-T Fusion Reactor with SS 316 Structure ($km^3_{water} \cdot sec/kW(th)$)

	IBHP
Fusion [b]	$5 \cdot 10^5$
Fission [c]	$6 \cdot 10^6$

[a] After 2 years of operation;

[b] T_2 removed after 10^6 sec;

[c] Reprocessing of actinides after 200 days.

In conclusion, it appears that, no matter how one compares the *inventory* of radioisotopes in fission and fusion reactors, fusion has a one to four order-of-magnitude advantage over fission, depending on the time of consideration.

5. CONCLUSIONS

It is evident that both fast breeder and D-T fusion reactors will contain high inventories (1000 to 5000 Ci/kW(th)) of radioisotopes after a few months of operation. Furthermore, the finite limits on fuel burn-up in fission reactors and finite blanket structure lifetimes in fusion reactors will result in large volumes of high level waste (10 to 100 m^3 per GW(th)·yr). This waste must be properly processed and stored for periods of several thousand years for near-term (stainless steel) fusion reactors, and at least ten times longer for fission reactors. More precise numbers require more precise criteria. The length of surveillance can be shortened to less than 50 years in fusion reactors if certain alloys of vanadium are used. However, the probability of this element being used in early fusion reactors seems small, because very little is known about its properties in a fusion reactor environment, and no commercial industry presently exists to produce the thousand metric ton quantities that would be required for an early fusion economy.

It is also recognized that the inventories of radioactivity in fission and fusion reactors, measured in curies, do *not* provide an adequate basis on which to compare the relative hazards. A somewhat better approach (although still imperfect because it does not include the pathways to release--see Chapter VII) is to use the biological hazard potential (BHP). The BHP index incorporates the effects of radioisotopes on humans either

by inhalation from the air or by ingestion through the water route. The inhalation BHP is mainly applicable in the event of an accidental release of radioisotopes either from the reactor or during any of the reprocessing steps before they are inserted into the final waste storage location. The ingestion BHP is mainly applicable to the long term storage of wastes but could also be important in specific accidental releases. On the basis of *inventory* alone (disregarding the probability of release, which is treated in chapters VI and VII), we find that the BHP for inhalation is one to two orders of magnitude higher for the LMFBR system than for fusion up to the point of the reprocessing of fuel wastes. After most (99%) of the Pu isotopes have been separated from the spent fuel, we find that the BHP for ingestion of the *inventory* (not the amount that would necessarily be released) is a factor of two to 100 times higher for fission than fusion for the first 1000 years after shut-down. For the next million years, the BHP for ingestion of the stored wastes is two orders of magnitude higher for fission than the structural material of a stainless-steel fusion reactor.

The economic incentive to reprocess fission fuels soon after discharge in large reprocessing facilities, serving about 10 to 30 GW(e) FBR plant capacity, means that large amounts of high level wastes must be handled, transported, and eventually solidified for long-term storage. Fusion systems have an integrated fuel cycle (tritium separation); and after appropriate compaction, structural steel and other waste material can directly be stored as solids. This tends to reduce the potential for a release of radioisotopes to the environment and could lessen the hazard potential associated with the final transportation of fusion reactor wastes to the ultimate storage facilities.

Finally, there is the question of measuring the total burden of radioisotopes to society. If one considers the BHP integrated over the lifetime of the various isotopes, allowing for the fact that any given isotope could pass through several biological generations, one finds that stainless steel D-T fusion reactors represent a factor of ten smaller burden than an LMFBR on the basis of a unit of energy produced. The use of vanadium alloys could increase this advantage for fusion to two orders of magnitude. As mentioned above, the probability that such alloys can indeed be used in economic fusion power reactors is much smaller than for similar reactors with a steel structure. However, hopes remain that structural materials for fusion with even better activation properties than vanadium, such as perhaps titanium alloys, will eventually be shown to be feasible [V-35].

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VI. NORMAL OPERATING LOSSES AND EXPOSURES

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1. NORMAL OPERATING LOSSES FOR THE LMFBR AND ITS FUEL CYCLE

Under normal operating losses for the LMFBR we discuss the production of tritium, C14, the noble gases Kr85 and Xe133, I129 and I131, as well as actinides and other radioactive wastes which are produced during the normal operation of fast reactors or other parts of the fuel cycle.

1.1 Origin of Activities and Production Rates

Most of the radioactive species that form a potential health hazard to the environment are either constituents of the fuel such as plutonium, or are generated by the fission process.

Tritium, however, is not only formed by ternary fission but is also generated by the $(n,2\alpha)$ reaction upon B10 that is contained in the control rods; a minor contribution comes from the (n,α) reaction upon Li6, an impurity in the coolant. C14 is generated by reactions upon impurities in fuel and cladding, mainly via the (n,p) reaction upon N14.

1.2 Release from the Reactor Plant

Tritium can diffuse through stainless steel and thus reach the sodium coolant. The tritium release from the fuel through the steel cladding is estimated at 50 to 90 per cent [VI-1 to VI-3]. However, tritium is chemically bound by sodium as sodium hydride, and then trapped by freezing in the sodium purifying system (cold trapping). Thus only a small share of the tritium generated in the reactor, amounting to something on the order of one per cent, can escape to the environment. It is conservatively estimated that 350 Ci/GW(e)·yr each goes into the atmosphere and the liquid effluents. The rest of the order of 25 kCi/GW(e)·yr is bound as Na-hydride waste in the sodium purifying system (cold traps); it is put to waste storage, together with other sodium impurities and corrosion products.

It is presently not expected that the release of C14 from the reactor plant will be in significant quantities.

The noble gases Kr85 and Xe133 can reach the cover gas of the primary circuits of an LMFBR plant only if some of the fuel rods (<1‰)have failed. A rod failure below 0.1 per cent is still considered normal operation.

The noble gases finally enter the cover gas purification system, where they are absorbed in one or more charcoal beds. This absorption effectively delays the flow of the noble gases of the order of days, the overall delay time for Xe being more than ten times larger than that for krypton. The noble gases leaving the charcoal beds may be separated from the cover gas by fractional distillation and collected, finally to be either bottled or released to the atmosphere under strict observation of the regulations for permissible radioactive releases.

In addition to this controlled purification of the cover gas, leakages of noble gases from the primary system must also be taken into account. Provisions may also be made for delays on this path.

For the reactor stations presently under construction, releases are estimated to be of the order of 300 Ci/GW(e)·yr for Kr85, and about 4000 Ci/GW(e)·h for Xe133 [VI-4]. It is anticipated that with further refinements they can be reduced by a factor 1000 or more [VI-1].

Iodine isotopes are important for the man-food cycle; here I129 and I131 have to be considered. These isotopes may be released from failed fuel rods. However, iodine then reacts with sodium to become NaI, which is trapped by freezing in the cold trapes of the sodium purification system. Small amounts of it can reach the cover gas of the primary-coolant system and escape to the environment, following its collection in storage tanks for controlled release. While the release of I129 may be neglected, I131 releases can be of the order of 10^{-2} Ci/GW(e)·yr [VI-3].

The amounts of plutonium and of other actinides that leave a reactor plant under normal operating conditions are negligible, compared to the emission of actinides by reprocessing and fabrication plants.

Table VI-I shows typical values of annual production rates for major non-fuel nuclides. Table VI-II gives a summary of the main radioactivity releases from a reactor power plant during normal operation [after VI-5].

Table VI-I: Typical Values of Annual Production Rates for Non-fuel Nuclides of Major Concern With Respect to Routine Emissions

	Half-life	Production Rate (Ci/GW(e)·yr)
H3	12.3 yr	$3.8 \cdot 10^4$
C14	5736 yr	10
Kr85	10.8 yr	$1 \cdot 10^5$
Sr90	28.5 yr	$6 \cdot 10^5$
I129	$1.5 \cdot 10^7$ yr	0.5
I131	8.0 day	$1.7 \cdot 10^7$
Xe133	5.3 day	$3 \cdot 10^7$

Table VI-II: Released Radioactivity During Normal Operation of an LMFBR Power Plant

	Release Rate (Ci/GW(e)·yr)	
	Into the Atmosphere	Into Water
H3	350	350
Kr85	0.4 (300*)	-
Xe133	0.03 (4200*)	-
I131	0.01	0.01

* Conservative numbers for reactors being presently built.

1.3 Normal Operating Losses from the Reprocessing Plant

In the reprocessing plant the fuel elements are chopped into parts of two to three cm length, which are put into dissolvers filled with nitric acid. During the chopping process most of the volatile fission products are released. With proper treatment of the off-gas it is possible to avoid uncontrolled escape to the environment. The spent fuel is then dissolved in

nitric acid, putting the remaining fission products as well as the fuel isotopes and higher actinides into solution. It is assumed that during the dissolution process the C14 generated in the reactor in part forms CO_2 and in part organic compounds. The CO_2 and the volatile organic compounds also enter the off-gas.

Efforts are to be made to transform all the C14 available into CO_2 so that it be treated in a similar way as the noble gases (cryogenic distillation).

The noble gases, in particular Kr85, are separated through contact with a fluorocarbon solvent at low temperatures. Thus the activity due to the noble gases is easily reduced by a factor of ten, while a factor of more than one hundred seems achievable.

For iodine, a retention factor of about 200 is attainable with present filtering techniques. With improved filtering techniques [VI-6, and VI-7], or via the Iodox process, which leads to concentrated iodine waste, target retention factors of the order of 10^4 seem within reach [VI-8].

Most of the tritium (70 to 90 per cent) already escapes from the fuel rods while they are still in the reactor. The remainder is distributed over the liquid (65 per cent) and gaseous (25 per cent) effluents of the plant, and a minor portion (10 per cent) enters the solid waste.

The confinement of plutonium and other actinides, notably americium and curium, is one of the salient design features of a reprocessing plant. Target confinement factors for future large-scale reprocessing plants will be $2 \cdot 10^9$ [VI-1]. This may be demanding. On the basis of measurements by the Public Health Service at the NFS plant at West Valley, New York [VI-9], an approximate calculation can be made where the plant confinement factor for airborne plutonium is set at about $2 \cdot 10^9$ during a 28 day period of continuous fuel dissolving and reprocessing [VI-9]. Yet it may still be demanding to achieve a containment factor of the order of $2 \cdot 10^9$ for the routine operation of a large reprocessing plant.

Table VI-III (after [VI-5]) summarizes the presently anticipated radioactive releases of a reprocessing plant during normal operation.

1.4 Normal Operating Losses from the Fabrication Plant

Fission products, or activation products such as C14, or the Cm isotopes do not enter the fabrication plant. Its radioactive inventory, therefore, is limited to the nuclides contained in the fresh fuel. The estimate of the releases from a mixed-oxide plant is based on the transmission factors of

Table VI-III: Main Activities of Effluents from a Fuel Reprocessing Plant Related to the Generation of 1 GW(e)·yr (reprocessing one year after removal from reactor)

Nuclide	Confinement Factor (annual flow/ annual release)	Activity (Ci/GW(e)·yr) Gaseous Effluents	Liquid Effluents
H3 (β)	1.1	$3 \cdot 10^{3}$	$8 \cdot 10^{3}$
C14 (β)	10	$5 \cdot 10^{-1}$	
Kr85 (β)	10	$1.6 \cdot 10^{4}$	
I129 (β)	$2 \cdot 10^{2}$	$3.5 \cdot 10^{-3}$	
Pu238 (α)	$2 \cdot 10^{9}$	$1.2 \cdot 10^{-4}$	
Pu239 (α)	$2 \cdot 10^{9}$	$3.5 \cdot 10^{-5}$	
Pu240 (α)	$2 \cdot 10^{9}$	$4 \cdot 10^{-5}$	
Pu241 (β)	$2 \cdot 10^{9}$	$4.8 \cdot 10^{-3}$	
Pu242 (α)	$2 \cdot 10^{9}$	$1 \cdot 10^{-7}$	
Am241 (α)	$2 \cdot 10^{9}$	$2 \cdot 10^{-5}$	
Am243 (α)	$2 \cdot 10^{9}$	$0.5 \cdot 10^{-6}$	
Cm242 (α)	$2 \cdot 10^{9}$	$1.4 \cdot 10^{-4}$	
Cm244 (α)	$2 \cdot 10^{9}$	$1.5 \cdot 10^{-5}$	
Total α Activity		$3.7 \cdot 10^{-4}$	

HEPA filters of 10^{-9}. If it is further assumed that 0.1 per cent of the plant's throughput becomes airborne, both numbers would combine to a confinement factor of 10^{12}. This is in part due to the less aggressive atmosphere in a fabrication plant. As the amounts of fuel throughputs per GW(e)·yr are similar in both reprocessing and fabrication, one may conclude that the radioactive releases from a fuel fabrication plant to the environment are comparatively low. But one should not overlook other pathways for the plutonium to escape. For instance, Pu-contaminated scrap must be handled with extreme care if a confinement factor of 10^{12} is to be maintained for the fabrication plant as a whole. Nevertheless, confinement factors of $4 \cdot 10^{10}$ have already experimentally been confirmed in pilot plants in operation [VI-10]. Blaylock et al. reported confinement factors of $1.3 \cdot 10^{11}$ [VI-11], whereas in the ERDA report WASH-1535 [VI-1] a confinement factor of 10^{12} is assumed. A confinement factor of $2.5 \cdot 10^{10}$ for Pu release to

water is reported for a 300 t/yr fabrication plant [VI-10]. For the purposes of this paper we assume that an overall confinement factor of $2 \cdot 10^{10}$ can be attained.

For the uranium-oxide fabrication plant (blanket elements), target confinement factors of 10^6 are sufficient. Confinement factors of $5 \cdot 10^5$ for U releases to the atmosphere were reported [VI-1]. Detailed data from the German fuel-fabrication plant RBU show confinement factors of $1.1 \cdot 10^5$ [VI-12]. Target confinement factors of 10^6 are within reach.

Table VI-IV gives a summary of the α activities expected to be released from a fuel fabrication plant [VI-5].

Table VI-IV: Activities Due to Releases from a Fabrication Plant

	Gaseous Effluents		Liquid Effluents	
	Confinement Factor	Activity (Ci/GW(e)·yr)	Confinement Factor	Activity (Ci/GW(e)·yr)
U isotopes	10^6	$8 \cdot 10^{-6}$	$2 \cdot 10^4$	$4.5 \cdot 10^{-4}$
α-emitting Pu isotopes	$2 \cdot 10^{10}$	$1.9 \cdot 10^{-5}$	$2 \cdot 10^{10}$	$1.9 \cdot 10^{-5}$
Pu241 (β)	$2 \cdot 10^{10}$	$4.8 \cdot 10^{-4}$	$2 \cdot 10^{10}$	$4.8 \cdot 10^{-4}$
Am241 + Am243	$2 \cdot 10^{10}$	$6 \cdot 10^{-9}$	$2 \cdot 10^{10}$	$6 \cdot 10^{-9}$
Total α-activity		$2.7 \cdot 10^{-5}$		$4.7 \cdot 10^{-4}$

1.5 Releases in the Fuel Cycle as a Whole

Figure VI-1 summarizes the assumed yearly releases per GW(e) in the fuel cycle as a whole. Contrary to the anticipated fusion-reactor designs, the technical installations for the fuel cycle are expected to be dispersed and not to be concentrated in one place. Later on (Chapter VII-1) we will elaborate further on this point, as this does not need to be the case.

It is now important to balance the releases given in Figure VI-1 against a yardstick. The US Environmental Protection Agency (EPA) has proposed to restrict releases to the following levels [VI-13]:

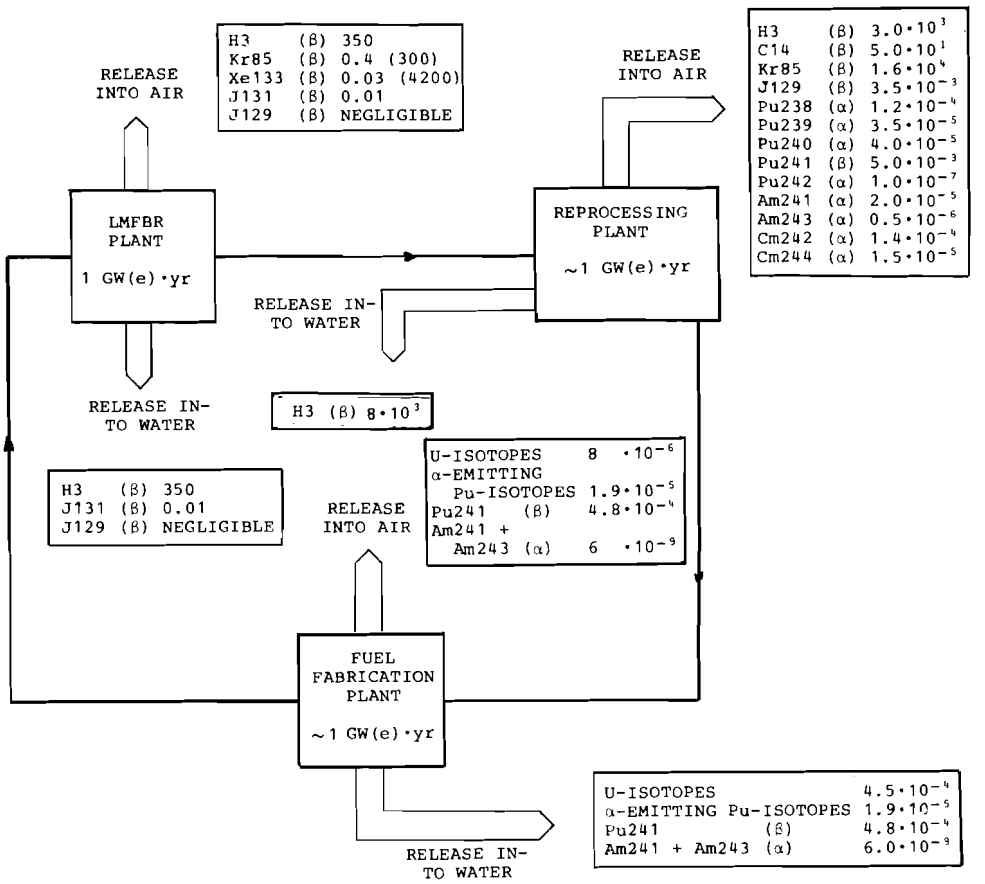

Figure VI-1: Operational Releases of an LMFBR Fuel Cycle (Ci/GW(e)·yr)

Kr85	50,000	Ci/GW(e)·yr
I129	5	mCi/GW(e)·yr
α emitters combined of Pu239 and other α-emitting transuranium isotopes	0.5	mCi/GW(e)·yr

Although these standards are still under debate, let us compare them with the numbers of Figure VI-1.

The reprocessing facility holds the greatest share. With the assumed confinement factor of $2 \cdot 10^9$ its releases are about 0.4 mCi/GW(e)·yr, and the contributions from the other facilities are insignificant. For iodine 129, the assumed confinement factor of 10^4 permits to stay well within the limit of 5 mCi/GW(e)·yr. I129 releases from the LMFBR plant are negligible. The confinement factor of 10 assumed for krypton puts the effluents below the contemplated limit of 50,000 Ci/GW(e)·yr.

It is presently debated whether the criteria for the Pu fuel cycle should be expressed not in terms of releases but in terms of whole-body dose rates of radiation, and figures of 25 mrem/yr are mentioned. This then includes consideration of the pathways in the analysis. For plutonium and iodine, this would not be simply an easy calculation of atmospheric dilution. We refrain from doing this in this paper, but what has been said illustrates how deeply such regulatory considerations can influence the design of fuel-cycle facilities and the build-up of the entire fuel cycle.

2. ROUTINE RELEASES OF RADIOACTIVITY FROM FUSION REACTORS

2.1 Tritium

The principal radioactive substance that will be released during routine operation of fusion power plants is tritium. As noted above, the inventory of tritium in the most recent TOKAMAK conceptual fusion-reactor designs is on the order of 10 kg (= 10^8 Ci) per 10^3 MW(th). This very considerable inventory, combined with the propensity of tritium gas to diffuse through metals and the difficulty of separating tritiated water from ordinary water, poses a non-negligible problem to fusion-reactor designers. In this subsection we briefly review the relevant physical and biological characteristics of tritium, and mention some of the technological approaches that are being considered to minimize routine tritium emissions in fusion-reactor operation. The question of larger, accidental releases of tritium is taken up in Chapter VII.

Tritium (H3, commonly written T) disintegrates by β decay (E_{max} = 18.6 keV, $\bar{E}$ = 5.7 keV) with a half-life of 12.26 yr, yielding the stable daughter He3. Chemically, tritium behaves like hydrogen; it typically occurs in the forms HT (tritiated hydrogen gas) and HTO (tritiated water), less commonly as T_2 and T_2O, and it can replace ordinary hydrogen in the whole array of organic and inorganic compounds containing that element. Because the mass ratio of tritium to hydrogen differs from unity by a large factor (in sharp contrast to pairs of isotopes of heavier elements), tritium is significantly discriminated for or against (as compared with hydrogen) in a variety of physical and biophysical processes (diffusion, vaporization,

uptake into organic compounds, etc.). Nevertheless, the *net* effect of such discrimination in most biological systems appears to be small.

In general, then, the fate of tritium in the environment can be rather accurately predicted by assuming that it goes where hydrogen goes, either as the gas or as the oxide. There is no evidence that T is concentrated in food chains; in other words, the ratio of grams of T to grams of H does not increase from lower trophic levels to higher ones [VI-14]. There are, however, some data for small mammals indicating that the T/H ratio in hydrogen that is incorporated in organic compounds in tissue exceeds the ratio in tissue-free water by about 50 per cent. Extrapolation of these results to man would suggest that the models currently used by the responsible national and international bodies may yield body burdens and concomitant doses from T, in a situation of continuous low-level exposure, that are too low by a factor of perhaps 1.8 [VI-14].

The MPC for tritium reflects the fact that it is far more dangerous in oxide than in elemental form. For HT and T_2, the MPC in air for continuous public exposure is $4 \cdot 10^{-5}$ Ci/m^3, as compared with $2 \cdot 10^{-7}$ Ci/m^3 for HTO. Elemental hydrogen gas is not retained when inhaled and is not absorbed through the skin to any significant extent. By contrast, water tends to be incorporated in body water when inhaled and is also readily absorbed through the skin. Intake of tritiated water from the environment also takes place through the pathways of drinking water and water contained in food.

Because of HTO's far greater accessibility to the body as compared with HT, it is customary to assume conservatively in hazard calculations that all tritium released is released as the oxide, or is quickly converted to the oxide in the environment. The amount by which this procedure overestimates the hazard in cases where the release actually is HT is not accurately known. Equilibrium constants indicate very high conversion of HT to HTO, both by oxidation and by exchange processes, but available data suggest that the rate constants are very small (on the order of days), even in the presence of catalysts [VI-14]. It is possible, then, that the hazards to humans *near the sources* of tritium that is released as hydrogen gas are overestimated by as much as two orders of magnitude by the approximation that it is all tritium oxide. In the present paper, however, all calculations are based on this conservative approximation.

Once tritium, incorporated in water, has entered the body, the ensuing hazard is governed by the biological half-life (the time scale on which it is excreted) and by the quality factor (QF) of its radiation. The biological half-life is given by M. Eisenbud [VI-15] as 9.5 ± 4.1 days, and the most widely used value seems to be 12 days. The International Commission on Radiological Protection (ICRP) dropped its value for the QF

of tritium from 1.7 to unity in 1969, and the value of unity is also used by the US National Council on Radiological Protection and Measurements (NCRP). Some evidence has been offered in support of higher values, but in no case for values greater than three [VI-14]. On the assumption of a QF of one and a biological half-life of 12 days, the incorporation of one Ci of T into the body of a "standard" adult human (mass = 70 kg) entails a whole-body-dose commitment of 72 rem. This is in agreement with a recent comprehensive review which puts forward a best estimate of 70 rem per Ci [VI-16]. Many studies report a larger dose, based on the old QF of 1.7 and the extremely conservative assumption that there is a "critical organ" for tritium consisting of 100 per cent water. (The published MPC for HTO in air is based on these conservative assumptions. That is, if continuous exposure to the MPC, accounting both for inhalation and for skin absorption, is assumed to produce a dose of 500 mrem/yr, then the corresponding dose commitment per Ci of intake is easily shown to be about 140 rem; this is just the figure one obtains by using QF equaling 1.7 and an organ of 100 per cent water.)

Experience with tritium to date has been associated mainly with its production in thermonuclear explosions (weapons tests) and in nuclear fission reactors. The data for fission reactors, in which tritium arises from ternary fission and from neutron interactions with light elements, provide a yardstick that lends some perspective to the likely figures for fusion reactors. A PWR produces on the order of 15 Ci of T per MW(e)·yr, hence 15,000 Ci/yr for a 1000 MW(e) plant at 100 per cent capacity factor [VI-17]. Of this amount, about 6000 Ci/yr would be discharged to the environment in the water leaving the plant's condensers, and the remainder (under present practices) would be discharged to air and water at the fuel-reprocessing plant. The inventory of T in a 1000 MW(e) PWR is about 20,000 Ci [VI-17]. Thus, the discharge rate at the power plant itself amounts to 30 per cent of the inventory per year, or about $1 \cdot 10^{-3}$ of the inventory per day. Presently envisioned technology for advanced fission reactors (e.g. the LMFBR) would reduce this discharge factor by one to two orders of magnitude, i.e., to between 10^{-4} and 10^{-5} of the reactor's inventory per day [VI-18, and VI-1].

To obtain an estimate of the degree of control over tritium that will be required in fusion reactors, let us assume that the predominant release pathway will be in the water discharged from the condensers. Assuming that all the rejected heat from a 2.5 GW(th) fusion-power plant operating at an 80 per cent load factor and 40 per cent thermal efficiency is carried off in once-through condenser water that is raised by 10^{0}C, the annual throughput of water is $9 \cdot 10^{8}$ m^{3}. If the requirement is that the HTO concentration could not produce a dose greater than 5 mrem/yr in an individual that takes all his or her drinking water from the condenser discharge, the maximum permissible annual discharge of HTO, based on 70 rem/Ci intake, is 10^{5} Ci.

This is about 300 Ci/day, or about one part in 10^6 per day (four parts per 10^4 per year) of the $2.5 \cdot 10^8$ Ci of tritium in the TOKAMAK conceptual design. This degree of control is three orders of magnitude tighter than that in today's LWRs, and one to two orders of magnitude tighter than that presently envisaged for the LMFBR. (If one bases the calculation in the fusion case on the mobile part of the inventory--the 10 per cent in the steam system--the permissible fractional discharge factor goes up by one order of magnitude.) Consideration of discharges to the atmosphere yields similar results: Emission of about 150,000 Ci of HTO per year from a 30 m stack would produce, under average meteorological conditions at a typical site, a dose of 5 mrem/yr at an exclusion distance of 600 m (based on 70 rem/Ci and meteorological calculations in [VI-19]). Here the required control factor is about six parts in 10^4 of the inventory per year, or slightly more than one part in 10^6 per day. If it were possible to guarantee that gaseous discharges were exclusively HT and T_2 instead of the oxides, this requirement would be relaxed about two orders of magnitude.

How will the needed degree of control in fusion reactors be attained? This question is receiving ever more attention in the increasingly detailed conceptual fusion reactor designs of various research groups [VI-14, VI-20 to VI-25].

There are two main pathways by which tritium in a fusion reactor could leak to the environment during normal operation. The first is through the heat exchangers into the steam system, from which the tritium can escape into the condenser-coolant water, and thus into the environment (as liquid HTO). The second is diffusion through the various containment-system boundaries, and eventual escape into the air around the plant as HT or gaseous HTO.

Typical fusion-reactor designs have dealt with the first of these pathways by making the fuel-extraction system (which removes tritium from the primary coolant) large enough and efficient enough to hold the T concentration in the primary coolant to very low levels; this low T concentration then limits the diffusion of T into the intermediate coolant loop (if any), and from there into the steam system. Design values for T release by this pathway on the order of one to two Ci per day per 1000 MW(th) have been reported [VI-19, and VI-20]. The second pathway generally has been regarded as being easier to control, and the resulting emissions have been predicted to be significantly smaller. The approaches used include separating hot T inventories from cold ones and making every design effort to minimize the former, surrounding hot T areas with cold metal walls, and employing copper, aluminum, or ceramic coatings as diffusion barriers.

The technologies envisioned to provide the extraordinary degree of tritium control required in fusion reactors remain to be proven in practice. How difficult and how expensive it will be to keep track of tens of kilograms of tritium to an

accuracy of one part in 10^6 per day in the real world of leaky valves, faulty seals, scratched diffusion barriers, and so on, will not really be known until we have tried it.

2.2 Routine Radiation Hazards Other Than Tritium

In addition to tritium, radiation hazards of possible consequence in the routine operation of fusion power plants come from the activation products in the reactor structure, the energetic fusion neutrons themselves, and the strong magnetic fields used to confine the fusion plasma.

It is widely supposed that activation products will chiefly remain in place in the structure until the activated parts are removed for reprocessing or disposal. Some of the metals involved form oxides that are much more volatile at operating temperatures than their elemental forms. The extent to which losses of these volatile oxides will contribute to radioactive releases from the plant and occupational exposures inside must still be investigated. A non-oxidizing atmosphere in contact with the activated structure might be necessary. One must note that, even when fixed solidly in place, activation products that emit penetrating γ rays pose a potential hazard to workers in and near the plant.

Since the objective in D-T fusion reactors is to harness the 14 MeV fusion neutrons for tritium breeding and thermal energy conversion, there is a strong economic incentive to minimize the escape of neutrons from the blanket region. It is also essential to protect the superconducting magnets from the neutron fluxes. Nevertheless, the great penetrating power of the fusion neutrons and the complex geometry of fusion-reactor interiors will make it difficult to completely shield the immediate area of the reactor. But sufficient shielding can be provided in the reactor building itself so that the neutron dose *outside* can be reduced to any desired level; thus neutron exposure will be an occupational hazard, but presumably not a public one.

Magnets for typical TOKAMAK reactors would produce magnetic fields that extend far beyond the reactor--500 m to drop to 1 G in UWMAK-I, for example. Prolonged exposure of fusion workers to magnetic fields of some tens of G is to be expected, and brief exposures to much higher fields are possible. The physiological consequences of prolonged exposure of humans to strong magnetic fields are not known; at routine exposures in fusion such effects may be non-existent, or negligible, or significant, and more research is needed [VI-14]. At some, possibly significant, expense the magnetic-field intensity outside the reactor building could be reduced by means of shielding with magnetic materials, or by partial cancellation with additional magnets.

3. CONCLUSIONS

Because the inventory of tritium is likely to be around 250 MCi/GW(e) in fusion compared to around 0.025 MCi/GW(e) in the LMFBR, the degree of tritium control would have to be about four orders of magnitude tighter in fusion to meet the same requirement on environmental doses from tritium (achieving 5 mrem/yr at the fencepost would mean tritium control to about 1 part in 10^4 per year for a 1 GW(e) fusion plant).

For the case of the fission breeder, the limiting factor is the release of α-emitters, iodine 129, and krypton. In all cases the reprocessing facility appears to make the largest contribution. Confinement factors (annual flow/annual release) of $2 \cdot 10^9$ for transuranium emitters, $2 \cdot 10^4$ for iodine, and 10 for krypton would permit meeting the regulations now under consideration in the US. As long as confinement factors of 10^{10} can be achieved, the fuel fabrication facilities do not seem to contribute significantly to the overall releases. The fast reactor itself is a small contributor compared to both reprocessing plants and fuel fabrication plants. All these confinement factors appear to be within reach.

Comparing fusion and fission with respect to routine releases, the degree of control required in the most sensitive part of the fuel cycle (tritium in fusion reactors, transuranium α-emitters and iodine 129 in fission-fuel reprocessing plants) appears to be attainable, but it still must be demonstrated in daily operation of large facilities. It also remains to be seen in both cases what the cost burden associated with these controls will be.

It is important to recognize the magnitude of the impact on technology of regulations concerning releases from the fast-breeder fuel cycle. It is necessary to define clearly the nature of such regulations and the specific levels that will have to be met, so that the technologists can adjust their designs accordingly. This illustrates a more general observation. While originally the inherent technical characteristics of nuclear power shaped the development of the technology, it is now more and more also the nature of regulations and standards which is shaping its development.

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VII. NON-ROUTINE RELEASES

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1. SAFETY OF FAST BREEDERS

1.1 Introduction

It is important to realize that up to now there have been three fairly distinct phases of dealing with the safety aspect of fast breeders. The debate within a broader scientific public and the general public is sometimes confused by not sufficiently distinguishing the origin of arguments from these three phases.

The first phase may be viewed to cover the years from 1944 to 1959. During this period much attention was given to the implications of a short neutron lifetime, the relatively small fraction of delayed neutrons, the delay time and reaction speed of control devices, the mechanism of a positive power coefficient, and the interaction of these features. The emphasis on these factors was enhanced by the fact that all of the considered breeder reactors were small in volume and power (not more than 66 MW(e)), and in particular by the fact that metallic fuel elements were used, which accentuated the apparent importance of criticality effects. (Metallic fuel elements were the best understood technology available during those years, and furthermore, it was felt that metallic fuel was a necessity to keep the breeding ratio sufficiently high.) Representative of this period were the design and operation of the EBR-I in Idaho and the EFFBR near Detroit (US), the DFR at Dounreay (UK), as well as the accompanying scientific and technological debate [VII-1].

At the end of the fifties the situation of the fast breeder development experienced a major change: mixed oxide fuel was considered as it inherently allowed much higher burn-ups, thereby guaranteeing a better fuel-cycle economy. At the same time the size of power stations (of any kind) went up in those years and, accordingly, the power output considered in fast-breeder design studies went up as well. By the middle of the sixties the size envisaged by reference design studies was clearly 1000 MW(e). This change then led to the second phase lasting until about 1970. During this phase the new dimensions for the design of fast-breeder reactors with ceramic fuel elements at 1000 MW(e) were

studied in depth in several countries of the world: the US, UK, USSR, France, the FRG, Belgium, Italy, and Japan [VII-2, VII-3]. An international community with strong interaction among the various groups developed the elements and concept of 1000 MW(e) fast-breeder power stations. A major part of that debate was the conception, identification, and analysis of the safety features of such a design.

Originally the debate around the size and sign of the Doppler coefficient of mixed oxide fuel was the focus of attention. This issue was resolved by the mid-sixties through studies in fast-neutron critical facilities such as ZEBRA, SNEAK, MASURCA, and ZPPR and, especially, in the 20 MW(th) fast-neutron test-reactor SEFOR (see Chapter III.1). SEFOR was designed, among other things, to measure the Doppler coefficient under normal operation conditions and transient power conditions [VII-4].

After the Doppler coefficient it was the sodium-void coefficient [VII-5] which attracted much attention. Sodium super-heat [VII-6], fuel/sodium interaction [VII-7], and fuel failure detection [VII-8] followed. At the end of that second phase a design concept arose from the international debate within the fast-breeder community that was satisfactory to the designers and solid enough as a base for the licensing and the construction of the 300 MW(e) prototypes in the USSR (BN 350), UK (PFR), France (PHENIX), and the FRG (SNR 300), with the US prototype (CRBR) and the Japanese prototype (MONJU) still to come. The licensing of the German SNR 300 was probably the most cumbersome of European prototype licensing since it had to undergo a full-fledged commercial licensing scrutiny. The difficulties of that licensing process were already indicative of what turned out to be the overriding feature of the third phase of fast breeder development. This third phase started around 1970 and is ongoing.

The overriding feature of the third phase is a widespread and multilevel approach of proofing the elements of the design concept that arose from the second phase. In this connection, one has to realize that the nature of what is considered a proof has evolved very substantially in the years since 1970. Engineering judgment was no longer considered sufficient. The probability approach to reactor safety was introduced and related to the concept of residual risks, extremely small in probability but very large in the perceived consequences [VII-9 to VII-11].

The focus of the scientific and technical attention of the third phase is, therefore, the *depth* and *credibility* of the experimental and theoretical *proof* of the safety concept of the present prototype and nearly commercial-size LMFBR plants. The point is *not* a change, a new physics element, or a major evolution of this safety concept as such. Since 1970 and, in particular, for the last two years many results of safety R&D-programs have been reported that are overwhelmingly in the reassuring direction. The sequences of events that underlie the safety concept of the

design of mixed-oxide fast-breeder reactors with up to 1300 MW(e) output are now a fairly complex matter. Contrary to the first phase of fast breeder development, simple lines of reasoning, therefore, no longer make up for the concept. Instead, the reasoning is now complex and largely refers to a considered design. This is the reason why it was pursued essentially within the fast-breeder community in close discussion with the licensing authorities. One must admit that this probably did not involve sufficient communication to the scientific public, which consequently led to what we consider misunderstandings.

In the following two subsections, we will present the safety concept of mixed-oxide fast-breeder reactors presently in operation or under construction as it evolved in the late sixties; we will then point to the strong experimental and theoretical proof supporting these designs which has piled up particularly during the last few years. Such a review is necessary because this accumulation of proof, which was done largely with strong international cooperation, has been almost unnoticed by the broad technical community and the public.

1.2 Dynamic Characteristics of Fast Breeders

Dynamic characteristics of fast breeders are characterized by a set of coefficients and constants that govern the related equations. These are the following:

- the effective fraction of delayed neutrons β_{eff};
- the effective decay constant of delayed neutrons λ;
- lifetime of the prompt neutrons l_{eff};
- reactivity coefficients, especially the Doppler coefficient, the fuel expansion coefficient, the structural coefficient, the coolant coefficient, and the sodium void coefficient.

The effective fraction of delayed neutrons from fissioning plutonium is 0.0035, which is half as much as the corresponding value in thermal reactors fissioning uranium 235. (This means that the dollar unit of reactivity, defined as the contribution of the delayed neutrons to reactivity. is itself half as large in Pu-burning LMFBRs as in U235-burning LWRs.) The relevant value, however, is always the ratio $\Delta k/\beta$ with Δk being the reactivity changes that are envisaged. In a fast-neutron spectrum, the Δk values in question are significantly smaller than in a thermal neutron spectrum, and the ratio $\Delta k/\beta$ is, therefore, equal to or smaller than that in the case of a thermal reactor [VII-12].

The prompt neutron lifetime in fast spectrum cores is about two orders of magnitude smaller than in thermal reactors (Table VII-I). This does not represent a disadvantage to the dynamic

behavior of fast breeders, however, as can be understood from the following explanation of subprompt-critical and superprompt-critical reactivity conditions for a fast reactor.

Table VII-I: Comparison of Reactor Kinetics Parameters for LWR and LMFBR [VII-14]

	l_{eff} (sec)	$\beta_{eff} \cong 1$ ($)	Decay Constants of Delayed Neutrons λ_i (sec^{-1})						
			Fuel	λ_1	λ_2	λ_3	λ_4	λ_5	λ_6
Pressurized Water Reactor	$2.5 \cdot 10^{-5}$	0.005 - 0.007	U235	0.0127	0.0317	0.116	0.311	1.40	3.88
SNR 300 Fast Breeder Reactor	$4.5 \cdot 10^{-7}$	0.0035	Pu239	0.0129	0.0311	0.137	0.331	1.26	3.21

Under subprompt-critical conditions, the kinetic behavior of the core is determined by the decay time of the delayed neutrons [VII-13 to VII-15]. As can be seen from Table VII-I, the decay constants for delayed neutrons of plutonium and uranium 235 are very similar. Therefore, the control behavior of fast-reactor cores under subprompt-critical conditions is not different from that of thermal reactors [VII-3, and VII-15]. The design of the control and shut-off system does not pose any technological difficulties, as can be seen from Table VII-II, where characteristic data of the control and shut-off systems are given.

Under superprompt-critical conditions $\Delta k/\beta$ becomes >1. It was shown by W. Häfele in 1963 [VII-16] that the short neutron lifetime, $l_{eff} \approx 3 \cdot 10^{-7}$ sec, is a fundamental safety problem only if the power/temperature coefficient is positive. With a negative power/temperature coefficient the opposite is true (other things being equal). The negative Doppler coefficients of PuO_2/UO_2-fueled fast cores leads to sharply limited narrow power bursts, until the delayed neutrons or the mechanical movement of the fuel (disassembly of the core) determine the time behavior [VII-12, and VII-16]. With a significant negative Doppler effect the energy release usually decreases with decreasing neutron lifetime [VII-17].

In addition, it has been shown by theory and SEFOR experiments [VII-18] that large mixed-oxide-fueled cores with more than 1000 MW(e) output have a strong negative power coefficient and a good control stability against reactivity or coolant-flow oscillations. In this case, the strong negative Doppler coefficient

Table VII-II: Comparison of Design Characteristics for Control and Shut-off Systems for PWR and LMFBR

		PWR (1300 MW(e))	Fast Breeder SNR 300
Speed of Control-Rod Movements	cm/sec 10^{-2}\$/sec	1 2.5	1.2 4.2
Speed of Shut-off Rod Movements	cm/sec	156	85-190
Delay Time Prior to Reaction of Shut-off System	sec	0.2	0.2
Time Span for Full Insertion of Shut-off Rods in Core	sec	2.5	0.5-0.7
Reactivity of Shut-off Rod System	Δk \$	11	10
Reactivity of Shim-rod of Boron System (Burn-up)	Δk \$	19	7

together with the negative structural and fuel expansion coefficients dominate the positive void coefficient [VII-12, and VII-19]. A positive fuel-element-bowing coefficient, as it was found in EBR-I in the early days of fast reactor design [VII-20], is always avoided now in large fast cores by the proper design of a steel-honeycomb-structure of the core with internal support plates and core restraint systems [VII-21].

In conclusion (and contrary to a widespread impression) one can state that the control behavior of fast reactors does not significantly differ from that of thermal reactors. The electro-mechanical design of the scram and control devices, therefore, is very similar to that of thermal reactors, as can be seen from Table VII-II.

1.3 Accident Chains

Accident chains for fast breeder reactors are considered on two levels:

(a) realistic accidents: Such accidents are known to be conceivable. Conception of such accidents does include chains of accidental events that are of low probability;

(b) hypothetical accidents: Conception of such accidents includes both

- chains of accidental events that are not proven to be inconceivable, and
- chains of accidental events that are ultimately conceivable but have a probability of occurrence that is below a predetermined level.

The problem of predetermining levels of probability for hypothetical accidents in the absence of the applicability of the trial and error approach has been dealt with in some depth by W. Häfele [VII-10]. It should be noted that hypothetical events are analyzed explicitly here (and in LMFBR licensing procedures in Western countries), and are not eliminated from consideration.

Realistic accidents are covered by the adequate choice of core safety parameters as well as by a proper design of instrument control of shut-down systems (Figure VII-1). Hypothetical accidents are covered by the strength of the primary cooling system (reactor vessel) and that of the surrounding containment system, as will be explained below. Safety analysis on the level of conceivable realistic accidents is not really the point of this paper, inasmuch as the concern of the public and the broad technical community is more with large hypothetical events.

We will, therefore, primarily discuss the safety questions of possible coolant blockages within fuel elements, the potential of fast reactors for core compaction, and the design-basis criteria.

1.4 Coolant Blockages and Subsequent Core Compaction

There is one line of events that is sometimes referred to as being hypothetical in the above sense. It is the following:

(a) coolant blockage in fuel elements;

(b) initiation of local boiling of sodium in the considered fuel element;

(c) damage propagation within the considered fuel element;

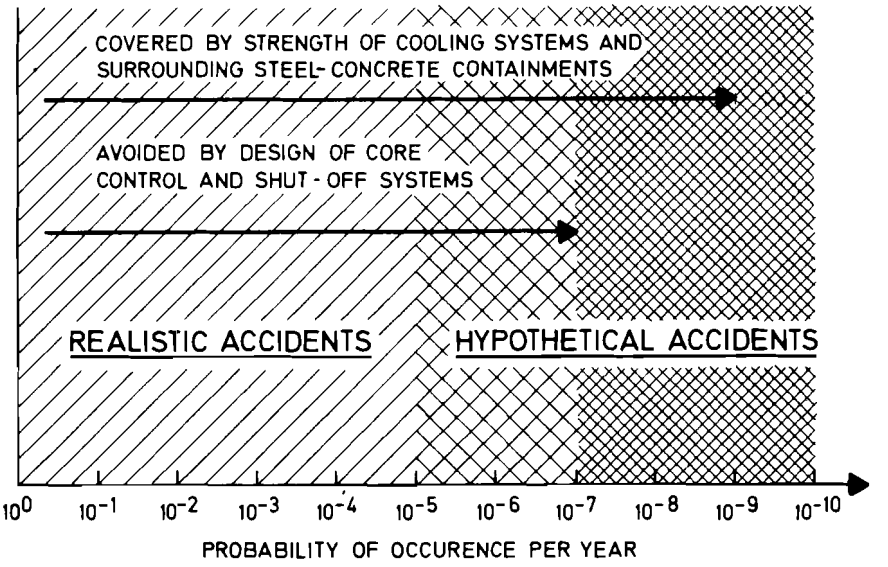


Figure VII-1: Realistic and Hypothetical Accidents

(d) integral gross boiling in the considered fuel element;

(e) melt-down of the fuel in the considered fuel element, and sodium-fuel interaction;

(f) pressure build-up, deformation of larger parts of the core, and coherent compaction.

There was the argument that if this sequence of events *(a)* through *(f)* could happen and, further, if this were to take place with precise coherence in a larger number or rather a complete ring of elements, this could lead to a major and disastrous energy release. This argument was put forward in particular by E. Teller [VII-22].

Experimental and Theoretical Evidence Against This Conceived Hypothetical Accident Chain

(a) Coolant Blockage in Fuel Elements

Local boiling can only be initiated if the coolant velocity is decreased by coolant blockages while the heat flux across the fuel rod surface remains constant. Two types of coolant blockages must be distinguished:

- blockage at the coolant inlet of the fuel element;
- blockage at grid spacers within the fuel element.

The first type of blockage happened in 1966 in the Enrico Fermi Fast Breeder Reactor [VII-23]. After blockage of the inlet of two fuel elements, a partial melt-down within the fuel elements occurred. No further damage was caused within the core, and the reactor was shut down safely. In present LMFBRs, this type of blockage can always be excluded by properly designing the fuel element inlet [VII-8, and VII-24].

For the second type of accident a continuous deposition of fuel particles and/or related debris at grid spacers within the fuel element is conceived. While this mechanism could be employed as an hypothesis, the phenomenon has not been observed so far during the operation of fast-breeder reactors like the DFR, BOR 60, EBR-II, PHENIX, or BN 350.

At Karlsruhe a thorough compilation and interpretation of in-pile experience on this issue has been made recently [VII-25]. Furthermore, the problem of depositing debris has been thoroughly simulated in out-of-pile experiments. At Karlsruhe a study [VII-26] compiled the results of such a simulation during 1974/75. A simulation subassembly was operated in which particles were added to the coolant stream. It was found that only large particles with sizes greater than 0.8 mm can be deposited and collected at grid spacers, where they could form coolant blockages. Smaller particles will always be swept through the fuel element. However, fuel particles being swept out of a failed fuel rod would be detected by delayed-neutron monitors before they could build up large blockages.

Such delayed-neutron detectors are capable of detecting damaged fuel surface areas as small as a few cm^2. This scheme was followed in particular by the French group during the start-up and operation of PHENIX [VII-27].

If inactive particles were deposited at the entrance of fuel elements, they would also be distributed uniformly so that the coolant flow would be steadily and slowly reduced and the outlet temperature of the fuel element would be detectably increased [VII-28].

Notwithstanding the evidence that even step *(a)* in the postulated accident chain is not a realistic possibility, we now consider, ad hoc, the next step.

(b) Initiation of Local Boiling of Sodium in the Considered Fuel Elements

At the end of the above mentioned second phase of the fast breeder development, that is the end of the sixties, local boiling was considered as part of a chain of conceivable events. One major piece of lacking knowledge was the percentage of the coolant cross-section to be blocked if boiling were to be expected. D. Kirsch [VII-29], in 1973, investigated that question

experimentally with an electrically heated fuel element, starting with full-scale elements and using water as coolant. A related computer code, developed by the UKAEA [VII-30] and applied to both UKAEA and Karlsruhe experiments, gave fairly good agreement [VII-31].

The major result is that even a blockage of 65% of the subchannels in the center of a fuel element will not initiate local boiling. For blockages at the fuel element walls these blockage sizes can be smaller. Final experiments to confirm this, on a 1:1 scale, are under way at Karlsruhe and Petten for the specific fuel element design of the SNR 300 [VII-32]. Similar results with different designs of fuel elements were obtained at ORNL [VII-33]. As discussed above, such local blockages of the fuel-element cross-section are extremely unlikely.

Only blockages of about >65% can lead to local boiling. We, therefore, consider the next step.

(c) Damage Propagation within the Considered Fuel Element

K. Gast at Karlsruhe studied the theoretical features of the propagation of damage originating from localized boiling [VII-8] in parallel with studies of H.K. Fauske (VII-34], and A.J. Brook [VII-35].

The important point of their conclusions was that sodium vapor bubbles originating from local boiling oscillate. Only if they oscillated slowly enough, thus permitting dry-out, would parts of the fuel cladding no longer be cooled, permitting damage to propagate. But this does not happen because the frequency at which vapor bubbles form and collapse is so high that the cladding area in question always stays wet. This important conclusion was experimentally confirmed in a simplified geometry by K. Schleisiek in 1974 [VII-36]. Final out-of-pile experiments in exact fuel-element geometry are underway [VII-32]. In-pile performance experiments to reconfirm these facts on a 1:1 scale and for specific fuel element designs are under way in Idaho in the ETR (SLSF-experiments), and at Mol in the BR-2 reactor (Mol-7C experiments). Even in the event that dry-out occurs and, as a consequence, fuel rods fail, one must keep in mind again the high sensitivity achieved in detecting the existence of small damaged cladding areas by means of delayed-neutron and cover gas monitors [VII-27].

We now consider, ad hoc, the next step.

(d) Integral Gross Boiling in the Considered Fuel Element

When the safety features of the present fast-breeder generation were analyzed at the end of the sixties, one mode of gross sodium movement initiated by gross boiling was of concern: coherent expelling and re-entering of sodium out of and into the fuel element (chugging). In the meantime, whole sets of experiments have been executed, in particular at Karlsruhe [VII-37], Grenoble [VII-38], Argonne [VII-39], O-arai [VII-40], and Ispra [VII-41]. The main result is that the expelled sodium does not re-enter the core region. Instead, there is chugging in the lower and upper blanket regions.

We now consider, ad hoc, the next step.

(e) Melt-down of the Fuel in the Considered Fuel Element and Sodium/Fuel Interaction

As in the above considered case, there were arguments at the end of the sixties to what extent hot fuel could suddenly interact with the coolant and thereby cause a vapor explosion. The first experiments that indicated that this was not going to happen were those of ANL made in the TREAT reactor [VII-42]. A surprisingly low thermal efficiency was observed. While very rough estimates with employing Carnot cycle models led to values as high as 10% to 20% [VII-7], the TREAT experiments indicated values lower than 0.2%. Thorough theoretical investigations by D.H. Cho [VII-43], L. Caldarola [VII-44], and others helped to understand these low values. Finite fuel/coolant mixing times, finite particle sizes, modes of fuel fragmentation and analyses of necessary conditions for explosive boiling [VII-45, and VII-46] were the main elements that were understood.

Along with these investigations one should include the evidence from out-of-pile experiments at ANL [VII-47], Grenoble [VII-48], and Ispra [VII-49]. Most of the informed experts conclude that we can assume conservatively an efficiency of only 1% to 2% of converting thermal energy into mechanical work.

We now consider, ad hoc, the next step.

(f) Pressure Build-up, Deformation of Larger Parts of the Core, and Coherent Compaction

During the late sixties concern was expressed within the fast breeder community that mechanical damage within one fuel element could propagate. Especially, it was E. Teller [VII-22] who expressed his concern about compaction modes for the core as a whole, following the sequence of events *(a)* through *(e)*.

In recent years such modes of interactions and possible propagations have been studied theoretically and experimentally. H. Jacobs [VII-50], in 1975, in greater detail studied the build-up

of pressure peaks in SNR 300 fuel elements. With extremely conservative assumptions peaks of 350 to 400 atm were calculated. Again the necessity was felt also to conduct experiments although they were very cumbersome, time-consuming, and expensive. A joint effort of the UKAEA and GfK Karlsruhe led to chemical explosion experiments in a 1:1 scale core model at Foulness [VII-51, and VII-52]. The major test series was conducted in 1974/75 for core materials with medium embrittlement simulating moderate burn-ups. Experiments with more brittle material will follow.

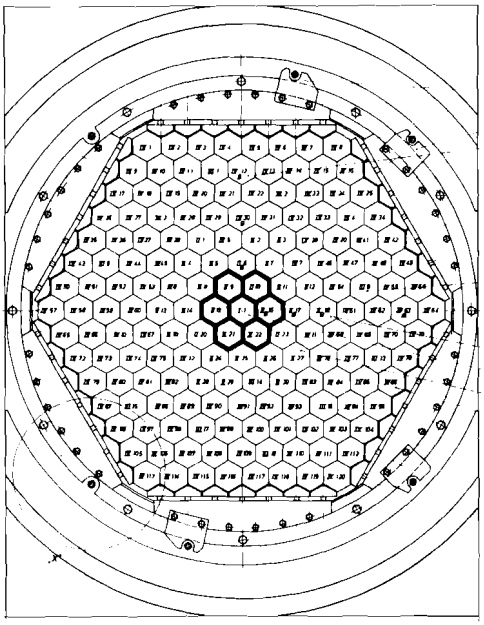
Informative results of the experiments performed so far are shown in Figure VII-2; they indicate the fact that the initiating exploding fuel element only compressed the neighboring fuel elements symmetrically by a few mm, without transferring this major deformation beyond the adjacent row of fuel elements; the deformation energy was absorbed there. This result appears to be essentially related to the hexagonal honeycomb structure and the plain fact that a fast-breeder core contains as much as 25% (by volume) of steel. All this happened at 720 atm, twice the conservatively calculated pressure build-up [VII-50]. The related reactivity effects were almost negligible.

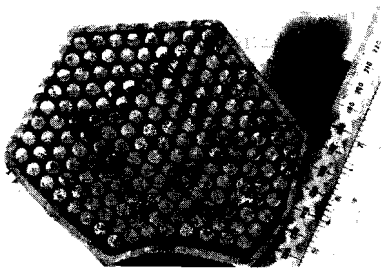
In addition, there is the point of extreme coherence in space and time that would be required if a gross compaction of the area as a whole were to be expected. The SNR 300 has 205 fuel elements; 1000 MW(e) reactors will have even more.

For inducing a hypothetical accident as conceived here, reactivity insertions of more than 2 $ are necessary [VII-53]. For such reactivity insertions to materialize through a chain of events *(a)* → *(e)*, not only the simultaneous occurrence of these events but also the above mentioned spatial coherence of compaction are required. The coherence scenario would have to be the following:

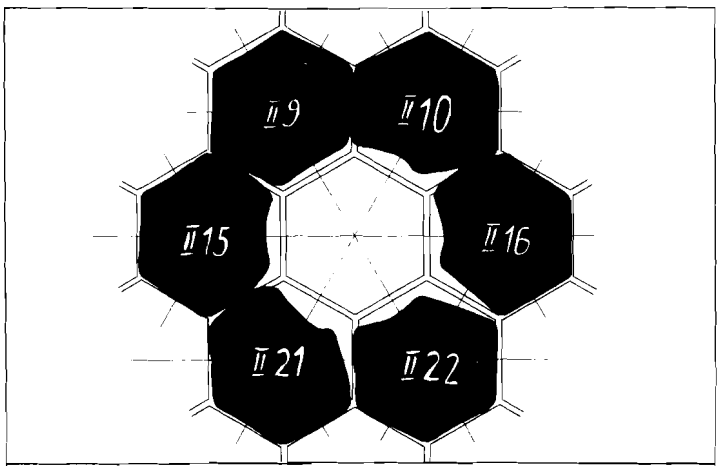
> At a given time local coolant blockages would have to develop in an outer ring of fuel elements of the fast-reactor core. In all these fuel elements, *time coherent* local sodium boiling, gross sodium boiling, fuel melting, and pressure development as a consequence of heat transfer from molten fuel to sodium would have to occur. The ultimate build-up of pressure waves would then have to occur *in all fuel elements coherently* in a msec time span.

Apart from the fact that a sequence of events *(a)* through *(e)* is not possible, as shown above, each of the different phenomena are in themselves statistical processes. The development of blockages and local boiling phenomena can last for either seconds or days, whereas the gross boiling and melt-down phenomena occur within several seconds. Sodium fuel interaction and pressure-generation phenomena would occur on a time scale of several msec. It is, therefore, concluded that a coherent pressure development in an outer ring of fuel elements of the core that follows a conceived sequence of events *(a)* through *(e)* is impossible.





SIDE FACING INCIDENT SUB-ASSEMBLY
SNR TEST 4
II 21



TEST 4: CROSS SECTION 1480mm FROM TOP OF ELEMENTS

Figure VII-2: Subassembly Deformation Tests by Internal Pressure Generation within One Subassembly of SNR 300 Core Structure (Maximum Pressure 720 atm within Reaction Zone)

Equally significant, however, is the fact that the experiments performed at Foulness have revealed the time scale of such conceived compaction mechanisms. Although deformation velocity of neighboring fuel elements after pressure generation in one blocked fuel element is in the order of a few 10 msec [VII-54], the next inner row of fuel elements would have to go again through the same conceived sequence of events *(a)* through *(e)*. The necessary time period is too long to generate large enough reactivity ramps or reactivity insertions.

It is, therefore, not surprising that the fast breeder community considers this hypothetical accident chain with events *(a)* through *(e)* followed by a coherent core compaction as now being known to be inconceivable. The events *(a)* through *(d)* are mainly discussed now for the specification of the in-core instrumentation system, with the aim of an early detection of cooling failures and fuel element damage in the core to increase its operational safety.

Instead, other sequences of events have led to the definition of the design basis criteria. The following chapter describes the sequence of considered events that led to the design basis criteria for the case of the SNR 300.

1.5 The Design Basis Criteria of the SNR 300

When discussing the design basis criteria of the SNR 300, as will be done in the following, it is important to observe that the SNR 300 is a typical case, and that essentially the same reasoning applies to other existing fast-breeder prototypes.

The SNR 300 has two independent shut-off systems that are not only redundant but also diverse in the sense that they are of completely different designs. One shut-off system operates from above, the other from underneath the core. For a large-scale hypothetical accident (core disassembly) to happen, it is assumed that both shut-off systems fail at the same time. If a failure probability of 10^{-4} per demand for each shut-off system is expected [VII-55], the conditioned probability for the two systems to fail simultaneously would be 10^{-8} per demand, assuming that there is no common-mode failure. Common-mode failures are avoided as much as possible by the use of independent design criteria for both shut-off systems. If, for any reason, there is an initiating ramp rate of a few \$/sec (up to five or so) and a simultaneous failure of the shut-off systems, then large fuel failure and, under certain conditions, sodium boiling will lead to the core-disassembly accident. It should be noted, however, that it is difficult to identify these initiating ramp rates in some realistic detail.

A pump coast-down is perhaps more easily conceivable. If pump coast-down occurs and the necessary simultaneous reaction of the two shut-off systems fails, this, too, leads to the core-

disassembly accident. First, there will be sodium boiling and then, depending on the size of the sodium void coefficient, fuel melting. In both accident scenarios that are considered here we must assume that there will be superprompt criticality. To achieve this in fact is not so easy. In this case, while the fuel is still essentially in its old geometry, there will be high fuel temperatures, and it is then critically important to understand the pressure build-up that goes along with this fuel temperature increase. The pressure/temperature relation is given by an equation of state. Throughout the various phases of fast breeder designs there has been a certain lack of knowledge. Thus it was only recently possible at EURATOM and in Japan to actually measure such an equation of state up to temperatures of 5000°C and above [VII-56, and VII-57]. This improved the confidence in this field and, as a main result, pressure/temperature relations that had been assumed during earlier years were confirmed.

The phase prior to disassembly has been modeled in the years 1972 up to now by various groups such as the ANL [VII-58] and Karlsruhe [VII-53]. It should be noted that these codes comprise neutron, temperature, and pressure fields together with the related material densities and governing power coefficients. In conjunction with these, detailed codes for the disassembly phase that follows pressure build-up in the fuel have been developed. They largely evolved from what formerly were known as Bethe-Tait calculations. For instance, the VENUS code of the ANL [VII-60] or the KADIS code of Karlsruhe [VII-61] deserve to be mentioned here.

Both code systems describe fuel heating, fuel failure, fragmentation, fuel coolant interaction, sodium boiling, fuel and clad melting, the pressure build-up of fuel and fission products , and mechanical movements in detail. Further, it should be noted that many experimental results are incorporated in these codes. These are in particular:

- Doppler and other power coefficients taken from SEFOR experiments under normal operating and superprompt-critical burst conditions [VII-18];
- TREAT and out-of-pile experiments for the fuel-sodium interaction, the mode of fuel failure, and sodium boiling as described above [VII-42];
- experimental confirmation of the equation of state.

The main results of this recent period of intense investigations for the case of the SNR 300 are the following [VII-53, and VII-62]:

- Nowhere in the core will the temperature rise beyond the 4000 to 5000°C level, and related pressures are below 150 atm. This result is significant, as early analyses of metal-fueled fast-reactor cores occasionally led to temperatures as high as 10,000°C and above and to pressure peaks as high as 10,000 atm.

- Together with the assumption of metallic fuel, this led to very high energy releases in these analyses. The main reasons for the recent results are the negative Doppler coefficient and the equation of state of the PuO_2/UO_2 mixed-oxide fuel.

- Not more than 5% of the fuel evaporates, 70% of the core is melted, while the rest of the core still has its original configuration.

- The energy release is 3000 to 4000 MWsec(th) above melting. For the SNR 300, the conservative assumption has been made that conversion of this thermal energy into mechanical work leads to a potential of 100 to 200 MWsec ($\int pdV$, p being pressure, and V meaning volume).

- Due to the size of the negative Doppler coefficient, there is considerable insensitivity against reactivity ramp rates [VII-12].

The SNR is designed in such a way that the inner reactor vessel and the primary cooling circuit can withstand 370 MWsec mechanical energy (design basis criterion; see Figure VII-3). Consequently, the necessary emergency cooling configuration can be maintained.

The proof of the integrity of the reactor vessel has been the subject of extensive theoretical and experimental proofing by groups such as the ANL 1965-1976 [VII-63], SRI [VII-64], NOL 1965 [VII-65], Foulness 1965-1976 [VII-66], Cadarache 1965-1976, Ispra, EURATOM 1968-1976 [VII-67], and INTERATOM 1970-1976 [VII-68, and VII-69]. High-temperature structural materials experimental programs are underway in the US and Europe [VII-70].

The core-cooling capability must be assured also for the time after a highly improbable core-disruptive or melt-down accident. Theoretical analysis for SNR 300 [VII-71] showed that a disassembled and destroyed core could very probably be sufficiently cooled within the reactor vessel, since the sodium remains in the reactor vessel; the primary cooling circuits as well as the emergency cooling systems would stay intact.

Recriticality conditions, if any, would hardly lead to prompt burst conditions, but more probably to smooth transient-overpower conditions, since the velocities for material displacements given by

- melting of insufficiently cooled parts of fuel elements, or
- fuel falling under sodium into subcritical assemblies of the core

are rather small. In addition, recriticality conditions would

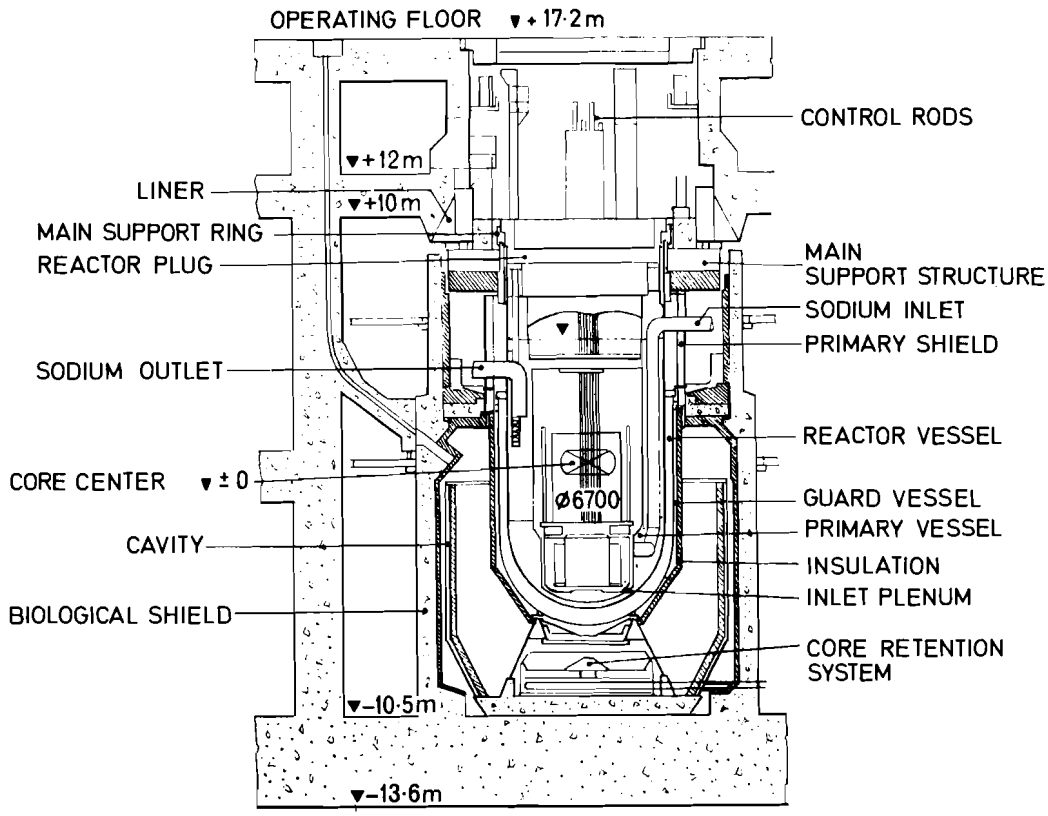


Figure VII-3: SNR 300 Reactor Vessel and Primary Containment System

heat up the sodium, increase its natural convection, and generate driving forces which could disassemble the fuel before it could reach prompt burst conditions. However, even if prompt burst conditions were attained, the possible consequences would again be covered by the design requirements for the reactor vessel and cooling system (design basis criteria).

In the case of the SNR 300, an external core catcher will be installed below the reactor vessel (see Figure VII-4). This additional cooling device must be understood as a second barrier of defense, in case molten core fuel should find its way from the core region through the core support plate and the bottom of the reactor vessel. It would then fall into the core catcher and be cooled there over a long period. A special geometrical arrangement of the core catcher avoids critical fuel configurations [VII-72]. The core catcher is cooled by a NaK system at its bottom and a nitrogen system at its sidewalls.

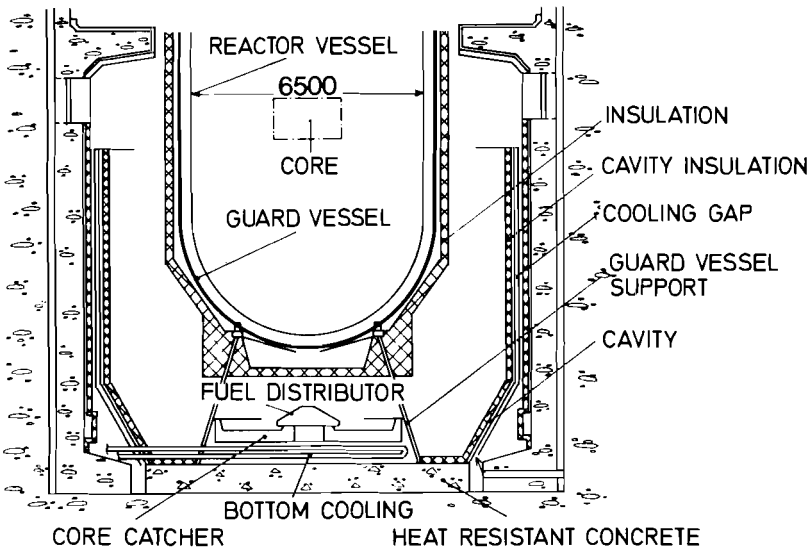


Figure VII-4: SNR 300 Core Catcher

Possible recriticality conditions within the core catcher were analyzed in [VII-73]. It has been found that, even under extremely conservative assumptions, the thermal energy releases during such recriticality accidents would be below those assumed for the design basis criteria.

Following the philosophy of defense in depth and in several layers, one can attribute safety functions to a double containment with prestressed concrete walls of up to 1.5 m thickness. There, one must realize that these containments have many varied functions (see Figure VII-5). The inner containment houses the heavy pieces of primary system components in a nitrogen atmosphere to avoid sodium fires, whereas the primary function of the outer containment is to establish the essential barrier for radioactivity and to protect the reactor from outside accidents such as airplane crashes. The outer containment of the SNR 300, for example, is designed against a Phantom-fighter crashing at a speed of 0.65 Mach against the containment walls [VII-74]. A particular function of the inner containment is to contain sodium aerosols while the secondary containment is, among other requirements, designed to contain aerosols from fuel handling accidents.

Given this situation one may in turn reconsider the potential of the containments for accidental releases of plutonium aerosols or aerosols of fission products and sodium.

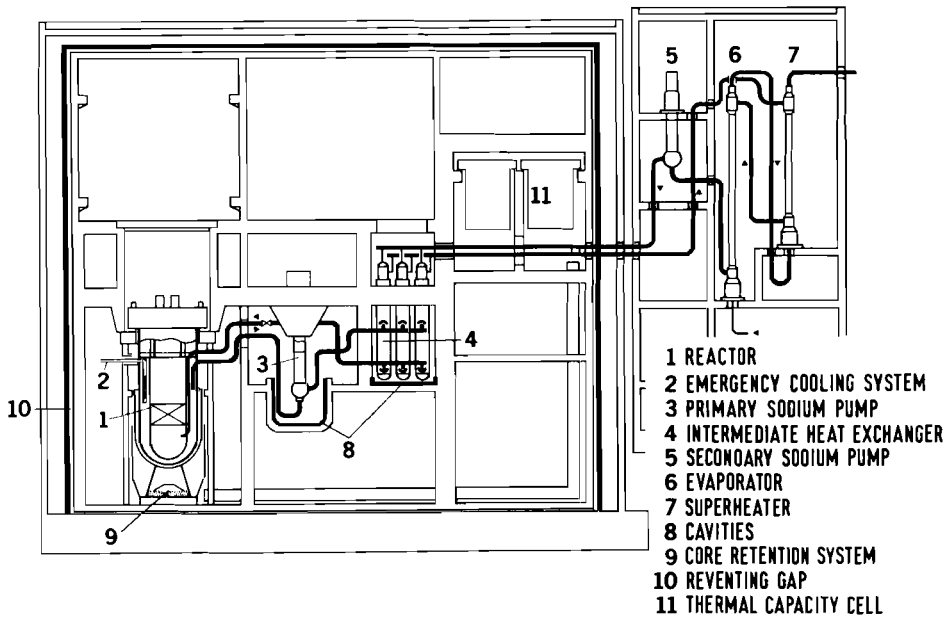


Figure VII-5: SNR 300 Containment System

Extended experimental work at BNL in 1965-1970 [VII-75], Karlsruhe in 1968-1972 [VII-76], Atomics International in 1965-1970 [VII-77], and Petten in 1968-1972 [VII-78], together with related theoretical work have led to computer codes like HAA-3 [VII-79], and PARDISEKO-III [VII-80], which allow description of the aerosol concentration in such containments as a function of time. In a follow-up of the above described defense strategy in depth one may, for example, assume a hypothetical event with releases of up to 60% of the core plutonium as fuel aerosols, and 100% of the volatile fission products and noble gases. Obviously, this assumption clearly contradicts the experimental and theoretical results, where the amount of core plutonium to be evaporated was only 5%. Given the present hypothetical event, in the inner containment of the SNR the following could happen: In part the aerosols would settle down by coagulation and sedimentation processes, but a small part of them could leak into the outer containment after some time. There the same coagulation and sedimentation processes would occur, but the aerosol leakages would be collected in the outer gas gap between the concrete wall and the outer steel shell and pumped back into the outer containment, which has a total volume of about 100,000 m^3. This reventing operation can be followed over several days until the aerosol concentration has reached an asymptotic value.

About 20 mg/m^3 of plutonium aerosols would be found to remain airborne. Only after these few days is it necessary to release some of the contents of the outer containment over sandbed and charcoal filters through a stack to the environment. Even in this extremely hypothetical event, which leaves the outer containment in order, the resulting irradiation dose at the fence of the reactor plant would be less than 1 rem total.

In a similar kind of ad hoc reasoning, one could bring into the picture the potential of the inner and outer prestressed concrete containments to withstand mechanical energy releases in the reactor cell containing the reactor vessel and the primary circuit. Although this potential is estimated to be significantly higher than that of the reactor vessel, it is not accounted for in the quantitative safety analyses since the vessel always stays intact as shown above.

The overall conclusion is that the LMFBR can meet the same predetermined safety standards as are applied to other reactors.

Before proceeding to the other topics of this Chapter, we must, for the sake of completeness, point to a somewhat alternative view of reactor safety in general. It is especially F.R. Farmer, UK, who rigorously stresses the probabilistic approach to safety in general; he embeds nuclear safety in the general problem of safety of technolocical facilities (e.g. chemical plants) [VII-81]. In particular, he points to the steadiness of the so-called Farmer curve, which relates increasing severity of accidents and decreasing related probabilities of occurence. Within such a framework anything can happen conceptually, and a distinction between realistic and hypothetical accidents, as favored, for instance, in the US and in the FRG, does not naturally derive from such a concept. Along with Farmer's approach there is an emphasis on accident prevention by early detection through sophisticated instrumentation that is based on a thorough understanding of the details of the technical systems and a deemphasis on protective measures, such as containment, in the case an accident has occurred.

1.6 Sodium Fires

As there is much concern especially about sodium fires a few remarks will be made here.

Sodium fires in the primary and secondary cooling circuits are prevented by embedding them into argon or nitrogen-filled cells. Extensive experimental data on sodium burning rates and burning temperatures, and sodium aerosol formation have been obtained from out-of-pile test rigs at Atomics International [VII-82], Cadarche [VII-83], and Karlsruhe [VII-84, and VII-85]. Computer codes such as SOFIRE [VII-86] and others are in use. These data together with the recently developed catch pan systems and fire extinguishing systems allow for a safe design of sodium systems. The operation of fast reactors like EBR-II, EFFBR,

RAPSODIE, PHENIX, PFR, and BN 350 have shown that the sodium technology has a high standard of experience.

In sodium water-steam generators, sodium may contact water after a pipe rupture. This problem has been investigated independently by all major fast reactor groups in the US, UK, France, FRG, Japan, and USSR [VII-87]. Extensive experimental data for temperature and pressure build-up and pressure-wave propagation through the steam generator bundle are available [VII-88].

Pressure relief systems have been designed which release the hydrogen formed during this Na-H_2O-reaction directly to the air. This does not lead to any radioactivity release since the sodium of the secondary coolant circuit is decoupled from primary sodium by an intermediate heat exchanger. Steam generator operational experience and incidents during start-up of the BN 350 and PFR were described earlier (Chapter III.1).

1.7 The Function of the Licensing Process

Given these explanations, one may come to ask why there is such a widespread debate about the safety of fast breeders. One reason for this was touched upon in the introduction to this chapter: the three phases of the fast reactor development have conveyed quite different kinds of concern to the scientific public, and often these historical views are mixed up.

But there is more to be said. The situation of safety of reactors in the very final analysis is to some extent open-ended. Experimental evidence related to safety analyses can only be given for partial aspects of the problem. A synthesis of those partial aspects for final overall assessments leads into a domain where the scheme of trial-and-error iterations can no longer be rigorously applied. This is in sharp contrast to the history of engineering. Safety analyses of boilers, for instance, did evolve from such a trial and error approach; engineers learned from accidents. The absence of the iterated trial-and-error approach therefore establishes some open-endedness also with respect to the related scientific and technical debate. It must be interjected here that this is not a special feature of nuclear engineering only. Indeed, all large-scale global engineering, in its consequences, leads to that situation. For instance, it is not possible to apply the iterated trial-and-error approach to the problem of climate changes on the globe. Nuclear engineering thus has assumed the role of a pathfinder for these modern global technologies and, more especially, it serves as as an instructive example of what has been labeled hypotheticality [VII-10].

In this situation the licensing process that was actually conducted serves to some extent as a mechanism for closing the open-endedness that otherwise exists. Assessments that will have consequences must be made. In greater detail, this means

that a synthesis of existing experimental and theoretical evidence is being made in the course of such a licensing process. This then provides a basis for further action in the field of construction but equally so in research and development. In the case of the SNR 300 a very comprehensive and lengthy commercial licensing process was experienced. In fact, to some extent it is still going on. It also leads to a considerable delay but must be judged as being extremely valuable. It turned out to be inherently as important as the original design phase of the SNR 300. A similar safety assessment was experienced with PHENIX in France and the PFR in the UK. On the other hand, for the American CRBR and the Japanese MONJU this licensing process is in its initial stages.

These soft procedural and institutional aspects may, to some extent, explain the ongoing public debate about the development of the LMFBR.

1.8 Safety Considerations for the Fuel Cycle of Fast Breeder Reactors and the Problem of its Spatial Deployment

In the last subsections, the safety of fast breeder reactors was considered in some detail. It was the reactor, of course, which received most attention. But for the commercially successful operation of fast breeders it is necessary to have also the various fuel cycle facilities established and in operation.

In the near and medium-term future, the overriding aspect for the FBR fuel cycle is the fact that the FBR accepts the Pu that is converted in the LWR, as both reactor types operate on the basis of the U-Pu-cycle. This LWR Pu is used for FBR first-core inventories; the timing of the LWR-Pu output, therefore, determines the timing of the commercial build-up of FBR. One has to realize that the hot end part of the LWR fuel cycle is being built up only now, after the installation of dozens and hundreds GW(e) of LWR capacity. It is, therefore, not surprising that related problems in a broader sense are only now receiving intense attention.

Figure VII-6 illustrates the fuel cycle of a fast breeder reactor on a steady-state basis. The fuel cycle comprises fuel pellet and element fabrication, and fuel reprocessing. After reprocessing, intermediate waste storage, waste solidification, and final waste storage must be considered.

If one wants to conceptualize the fuel-cycle problem as a whole and that of its safety in particular, one is led to a somewhat aggregated approach. Recently R. Avenhaus, W. Häfele, and P. McGrath did so with a view to ordering and categorizing safety and operation problems of a large fuel cycle for 1800 GW(th) FBR and 1800 GW(th) HTGR capacities [VII-89]. We make use here of that paper's findings and numerical results.

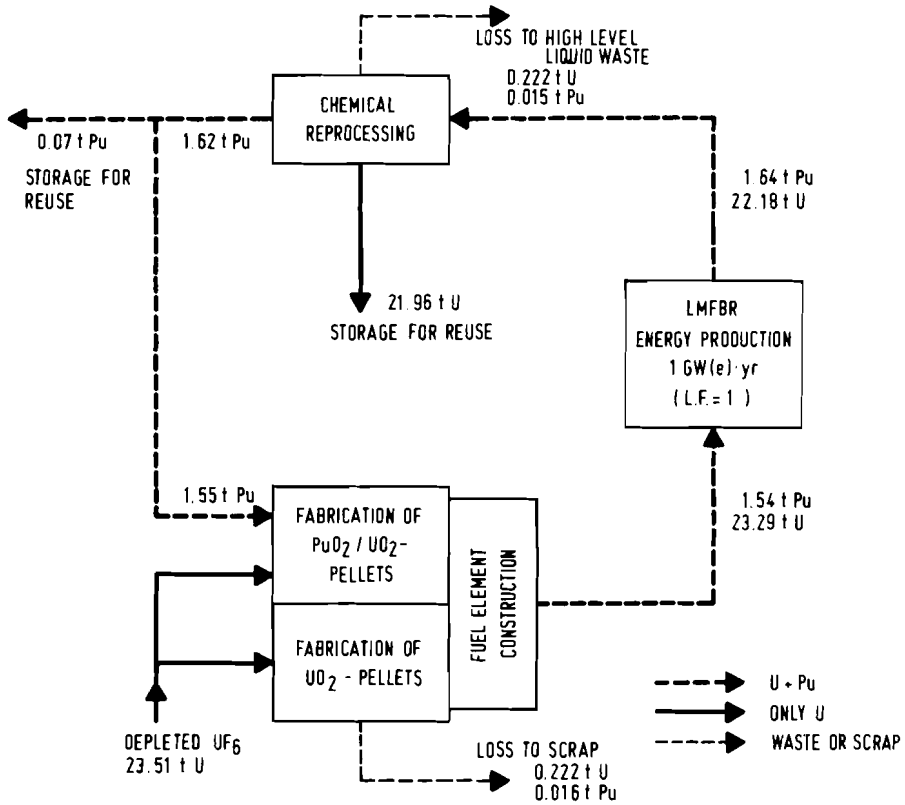


Figure VII-6: Integrated LMFBR Fuel Cycle System - Yearly Mass Flows of Uranium and Plutonium for 1 GW(e)

As regards the safety aspect of the problems, the approach of that paper is a normative one. It is assumed that a predetermined amount of radiation dose is accepted as a consequence of one or more accidents over the lifetime of an individual. This requires regulation and other legal steps. For the time being, the values of 25 rem per individual and a 70 year lifetime are employed. Further, the notion of expectation values is rigorously used. It means linear averages over small and large events and over time. Such rigorous application of the concept of expectation values has obvious shortcomings, but it is felt to facilitate a rough, first-order orientation and to identify more specifically the steps that would lead beyond the concept of expectation values.

This approach is described by the following simple relation (see Figure VII-7): B is the considered accidental dose rate of an individual assumed at 25,000/70 mrem/yr; BM is the population dose rate in manrem/yr; it is necessary to incorporate assump-

NORMAL OPERATIONAL LOSSES	DOSE RATE = EMISSION · METEOROLOGY / POPULATION · BIOLOGY $B \left[\frac{mrem}{yr}\right] = Q \left[\frac{Ci}{sec}\right] \cdot s \left[\frac{sec}{m^3}\right] \cdot \varrho \left[\frac{mrem/yr}{Ci/m^3}\right]$ $BM \left[\frac{manrem}{yr}\right] = Q \left[\frac{Ci}{sec}\right] \cdot \int s(r) \cdot f(r)\, dF \left[\frac{sec}{m^3}\right]\left[\frac{man}{km^2}\right]\left[km^2\right] \cdot \varrho \cdot 10^{-3} \left[\frac{rem/yr}{Ci/m^3}\right]$
FOR ACCIDENTAL LOSSES SUBSTITUTION OF	$Q \left[\frac{Ci}{sec}\right] \longrightarrow P \left[\frac{1}{sec}\right] \cdot \frac{d}{3.15 \cdot 10^7} \left[\frac{sec}{sec}\right] \cdot C \left[Ci\right]$

DEFINITIONS

B : ACCIDENTAL DOSE RATE FOR AN INDIVIDUAL: 25,000/70 (mrem/yr)
BM : POPULATION DOSE RATE
Q : RELEASE RATE
s : METEOROLOGICAL DILUTION
ρ : IMPACT FACTOR RELATING AMBIENT DOSE RATE TO DOSE RATE OF AN INDIVIDUAL
f(r): POPULATION DENSITY
F : CONSIDERED POPULATED AREA
P : PROBABILITY FOR AN ACCIDENT TO HAPPEN
C : RELEASE RATE
d : EXPOSURE TIME AFTER AN ACCIDENT: 50 h

Figure VII-7: Relations Between Dose Rates and Release Rates within the LMFBR Fuel Cycle

tions on the population density f(r) and the considered populated area F ; P is the probability per second for the considered accident to happen; d is the conceived exposure time after the accident, here assumed to be at 50 hours; C is the release in curies; s is meteorological dilution, and ρ is the factor that relates an ambient dose rate to the dose rate of an individual. It should be noted that ρ may be given by regulations of the International Commission on Radiological Protection (ICRP), s by the local meteorology, C by the technical accident scenario, d by postaccident scenarios, and B by (assumed) regulations. It is then possible to calculate P. If more than one isotope is considered, proper combinations of C•ρ can be used to describe

the situation accordingly. P must be interpreted as a demand for reliability; it is of a normative character. Small P values imply a more dangerous situation where accidents can be accepted only rarely (within the scope of the approach considered here). Large P values, in turn, imply more acceptable and thus less dangerous situations. Actual design basis accident probabilities of a given facility must then be smaller than these normative values. Reliability control studies like those by Rasmussen for the LWR case must sufficiently assure that this is so. In this format, the above mentioned study considers not only accidents in facilities but also the case of physical protection and adverse spreading of Pu. Table VII-III shows some of the results of that study. It is surprising to learn from the table that reprocessing, plutonium contamination, and the explosium of a crude device for which plutonium was obtained through inadequate physical protection, are of less concern than intermediate waste storage, the case of a Pu fuel fabrication plant, and the case of final waste disposal. It is unavoidable to make certain assumptions in all these rather simple and straightforward calculations, and these assumptions, of course, are subject to discussion. Table VII-III also gives relative (and absolute) values of population dose rates. This is done on the basis of assumptions on the population density as explained there.

Table VII-III: Normative Accidental Losses P_D^o for 3600 GW(th)

	Reprocessing	Intermediate Waste Storage	Fabrication Plant	Final Waste Storage	Contamination	Crude Explosive Device
$\frac{BM}{BM_o}$ [b]	$7.2 \cdot 10^{-5}$	$3 \cdot 10^{-3}$	$2.5 \cdot 10^{-5}$	0.15	$3.2 \cdot 10^{-4}$	0.02
$P_D^o \left[\frac{1}{yr}\right]$	0.4	$5 \cdot 10^{-3}$	$P_D^o \cdot X = 1.4 \cdot 10^{-3}$ [c]	$P_D^o \cdot F_1 \cdot F_2 = 4.2 \cdot 10^{-6}$ [d]	$P_D^o \cdot X = 14$ [c]	0.05

[a] For B = 25,000/70 (mrem/yr); B_o = 110 (mrem/yr)--natural background radiation exposure;

[b] $BM_o = 4 \cdot 10^7$ (manrem/yr); meteorology/population factor: $2.8 \cdot 10^{-4}$ (mansec/m^3);

[c] Amount of X (g) Pu released;

[d] F_1: percentage of waste cylinders exposed to water;
F_2: soil filtration factor.

A special case is final waste disposal, where the underlying assumptions heavily influence the resulting ordering of concerns. Figure VII-8 [VII-90] illustrates the scenario that was assumed Waste is stored in glass cylinders of 20 cm diameter, in which a break-in of groundwater occurs because of unforeseen geologic

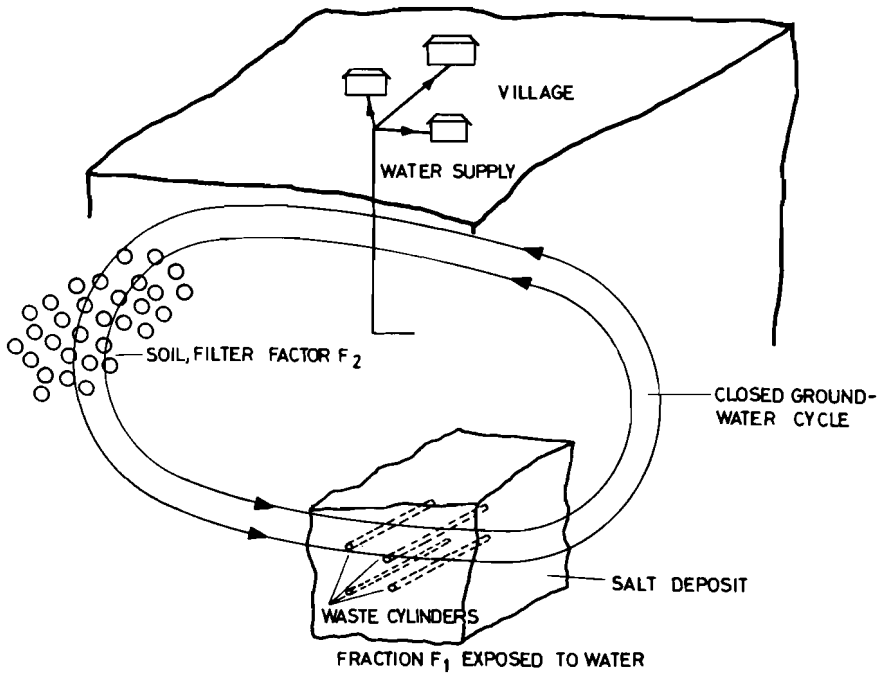


Figure VII-8: A Waste Disposal Accident Scenario

events. Fraction F_1 of the total glass cylinder surface is assumed to be exposed to water. It was further assumed, ad hoc, that the groundwater circulates in a closed loop. The soil filters the water, resulting in a filter factor F_2. Finally the assumption is made that people would have to use the groundwater for drinking for ten years, the period needed for appropriate measures to stop groundwater circulation by technical means. Given all these assumptions, the resulting normative accident probability is the one of Table VII-III.

These assumptions are fairly arbitrary. But this reasoning may help to point to the inherent necessity of scenarios for reference purposes. In fact, the design basis accident (DBA) related to the design basis criteria as discussed in the above subsections fulfills this function for the reactor. As the reactor accident is to be contained within the reactor building, all features of the scenario refer to technical equipment, measures or events within the reactor building and are, therefore, strictly within the scope of the reactor design work. In the case of final waste disposal, and to a lesser extent also for some of the other parts of the fuel cycle, such conceiving of scenarios cannot be confined to the interior of a building.

One must define a design basis scenario that does refer to environmental and geologic features of an actual case. Because of this very fact, it is not possible to draw compelling conclusions from this subsection. It is, nevertheless, felt that procedures for doing so have been outlined, and above, certain indications have been given by Table VII-III [VII-90].

The fuel cycle of fission reactors is spread out in space. This is a consequence of the law of scale. Chemical reprocessing, for instance, becomes commercially reasonable above the level of perhaps 1500 t/yr for one single plant. This throughput services about 45 GW(e) and, because of it, chemical reprocessing must be centralized. This is not always the case: the design of the EBR-II facility incorporates a closed fuel cycle, and in the case of the molten salt reactor the closed fuel cycle is an integral part of the concept.

The necessity of centralized reprocessing facilities and of a decentralized deployment of nuclear power stations relays a certain disparity within the nuclear power concept. Thus size of electrical power stations in general is a result of three competing influences: *(a)* the size of the grid which provides the scale of the power stations, *(b)* the striving for availability which tends to favor small power stations, and *(c)* law-of-scale incentives which tend to push for large power stations. Optimal sizes seem to be around 10% of the grid capacity. The grid capacity in turn is determined by the constraints of the transport of electric power. With today's and tomorrow's technology, the average distance of travel of one kWh is at 100 km and, accordingly, an optimal nuclear power station size is only at 1 GW(e) in contrast to the size of a reprocessing plant. If electricity cannot be transported the nuclear fuel, fresh or irradiated, must be transported.

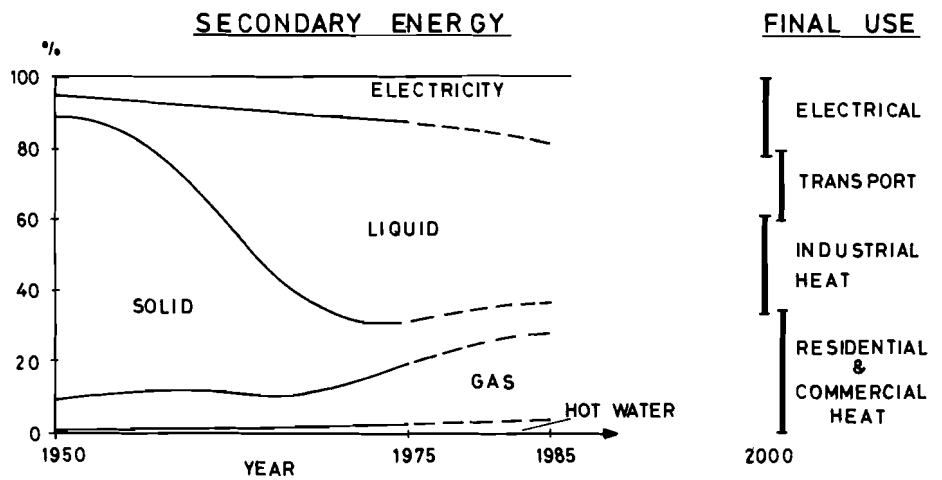


Figure VII-9: Partitioning and Final Use of Secondary Energy (FRG)

One must now realize that nuclear power does not necessarily have to produce electricity. In fact, the so-called energy problem rather asks for a non-electric, gaseous secondary energy carrier, as outlined in Figure VII-9 [VII-91]. Gasified coal, split methane [ADAM + EVA [VII-92]), ammonia and, above all, hydrogen are canditates for such a non-electric secondary energy carrier. What is important in this reasoning is the fact that a gaseous secondary energy carrier can be far more easily transported than electricity. While electricity transportation is de facto restricted to a few GW(e) and a few hundred km, the transportation of gases in pipelines is capable of transporting dozens of GW over several thousand kilometers [VII-93]. This observation now leads us to a natural partition of tasks: Normal nuclear power stations, based on U235 fueling and along the lines of the technical experience now available, would continue to function as electric power stations on a decentralized basis. The irradiated fuel elements would be transported to collocations, and the Pu would never leave these energy centers. Instead, it would fuel reactors that would probably produce mainly the process heat for synthesizing a gaseous secondary energy carrier with subsequent easy transportation in an (already existing?) pipeline system. Of course, we do not want to exclude electricity generation in such energy centers, where appropriate.

One arrives, therefore, at a sequence of modes for the geographical deployment of nuclear energy as shown in Figure VII-10 [VII-90]. As time, and with it, the capacity of a modern energy system evolves, we are now, after the transition from coal to oil,

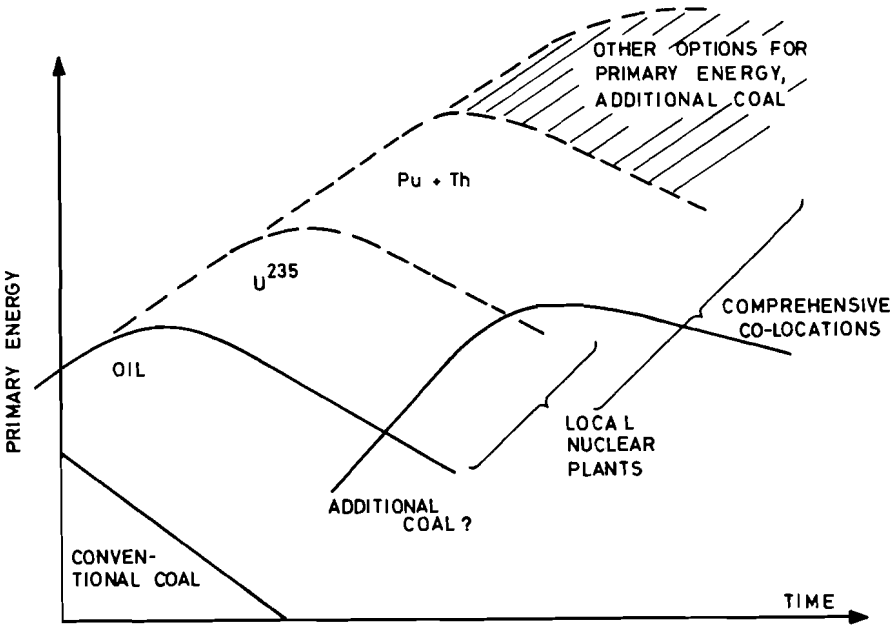


Figure VII-10: Modes for the Geographical Deployment of Nuclear Energy

beginning to see the transition to local nuclear plants on the basis of U235 fuel. The nuclear community now faces the problem of appropriate uses of large plutonium amounts. We here suggest as a further transition using this plutonium in comprehensive collocations, i.e. in large centralized energy centers.

The reasoning given throws a light upon nuclear fission power as a whole: It inherently fits with large-scale production and transportation; it fits better into the TW domain than into the GW domain. Thus nuclear fission power allows for collocations of nuclear facilities, partially or totally eliminating transportation. Judgments on the safety of the nuclear fuel cycle thus depend very strongly upon the patterns of land use, degree of centralization, and the underlying infrastructure for secondary energies. Contrary to the case of nuclear reactors, considerations cannot be restricted to the inside of containments.

2. NON-ROUTINE RELEASES--FUSION

This particular topic is very difficult to treat for fusion plants, because they have not actually gone through a complete and detailed engineering design. The best one can do at this time is to analyze the current point designs for general features that *could* pose problems. In all likelihood, many of these problems will be solved as the engineering design becomes more sophisticated. There have been five detailed attempts at environmental studies, for the UWMAK-I, UWMAK-II, and UWMAK-III [VII-94 to VII-95], LASL-ANL Theta Pinch [VII-97], and the BNL reactors [VII-98]. Recently a research group at the University of California, Los Angeles, has attempted an independent analysis of the UWMAK-I reactor. It is intended to combine their work with the work on UWMAK-II and UWMAK-III [VII-99], in order to get some qualitative picture of what to expect in a TOKAMAK design.

Preliminary analyses reveal that there are four main sources of hazard to the public outside the reactor. (Obvious hazards to the operating personnel inside the reactor, such as fire, explosions, steam-pipe ruptures, and other "conventional accidents", are not discussed here.)

The hazards to the public are:

- release of tritium;
- release of radioactive corrosion and sputtered products entrained in the coolant;
- release of radioactive structural and breeding material, neutron-multiplying, reflecting, shielding, electrical and organic materials;
- release of non-radioactive but toxic materials.

In what follows we first examine the accident pathways that could lead to the manifestation of these hazards, and then consider the potential consequences.

2.1 Accident Chains for Fusion Reactors

Without careful evaluation of more detailed reactor designs than those yet available, a detailed quantitative examination of accident probabilities for fusion reactors is not possible. Nevertheless, some useful observations can be made on the basis of the amounts of energy likely to be stored in various parts of fusion-reactor systems, and in the pathways potentially available for the release of such energy.

The amounts of energy stored in a TOKAMAK reactor based on recent conceptual designs are summarized in Table VII-IV.

Table VII-IV: Stored Energy in a 1000 MW(e) TOKAMAK Fusion Reactor

	Stored Energy (GJ)
Kinetic Energy in Plasma	0.1 to 3
Complete Fusion of Fuel in Plasma	70
Magnetic Energy	240
Energy Stored in Maintenance of Vacuum	16
Chemical Energy in Liquid Lithium	64,000*

A rapid release of the nuclear energy represented by the fuel contained in the plasma at any given time seems exceedingly unlikely on the basis of present knowledge of plasma behavior. It is conceivable that malfunctions could temporarily increase the reaction rate, by virtue of increased temperature or magnetic field, but the delicate balance of conditions required for the plasma containment means that the final result of any departures from normal operating conditions would be rapid quenching of the reaction, due to loss of plasma to the walls. If somehow the entire quantitiy of fuel in the plasma did react, the less than hundred gigajoules that evolved would raise the blanket temperature by only about 100^{0}C [VII-100].

The kinetic energy of the hot plasma is two to three orders of magnitude smaller than the potential nuclear energy. If an instability were to cause the entire hot plasma to be deposited on a small section of the vacuum wall, a local burn-through could result [VII-99, and VII-100]. Such an event would naturally be disruptive and expensive for the operators of the reactor, but it could only produce serious external consequences if it led to a major lithium fire, as discussed below.

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* This could be reduced by the use of solid breeders.

A very substantial quantity of energy, hundreds of GJ, is stored in the superconducting magnets that confine the fusion plasma. Concern is sometimes expressed that the sudden release of this energy, owing to a magnet transition from the superconducting to the normal state, would represent an important accident chain for fusion reactors, and considerable analysis of the question has been done [VII-99, and VII-101]. It seems a relatively straightforward matter, however, to design magnets incorporating normal conductors as alternate-current paths, along with adequate energy sinks for the associated joule heating, so that an explosive energy release from a magnet "going normal" is precluded [VII-101].

It will remain true, of course, that large superconducting magnets in operation are subject to very large forces (on the order of 10^5 t), so that catastrophic structural failure perhaps cannot be entirely ruled out. Such a failure could produce energetic missiles, raising the possibility of severe damage to other reactor components, penetration of containment structures, and initiation of lithium fires.

Energetic missiles, with the same possible consequences, could also be produced by sudden vacuum-system failure, by helium overpressure in the magnet-cooling system, by "external" events such as earthquakes, tornados, aircraft impact, and sabotage, and of course by combinations of these pathways [VII-99, and VII-94].

An important accident chain for all nuclear systems is loss of coolant, or loss of coolant flow. Calculations made for the UWMAK-I/TOKAMAK conceptual design indicate that complete loss of coolant flow during thermonuclear burn, accompanied by failure to shut down the fusion reaction, would cause the first wall to reach a temperature of 600°C in about 10 sec; embrittlement from formation of helium bubbles in the metal, which occurs around 650°C, could then lead to failure of the wall, release of lithium into the vacuum chamber, and consequent quenching of the fusion reaction [VII-102]. Like so many other fusion accident chains considered here, this one would be disruptive and expensive but not catastrophic unless secondary events produced a major fire and/or breach of containment.

Loss of the coolant itself (as opposed to mere loss of flow), as could occur owing to pipe breaks, would produce the same result as loss of flow somewhat more rapidly, again assuming that the fusion reaction were not immediately shut down. If, on the other hand, the reaction is shut down, the concern becomes the radioactive decay-heat that results from activation products in the first wall. The initial decay-heat power densities in the first wall at shut-down fall in the range of 0.5 to 1.0 W/cm^3 for the main alternative structural materials [VII-101], and the initial adiabatic temperature rise in a TOKAMAK system has been calculated to be on the order of

0.1°C/sec [VII-94]. These values are more than an order of magnitude lower than the corresponding figures for fission fuel, and they suggest that heat removal by radiation, conduction, and natural convection will suffice to prevent melting of the wall.

The largest source of stored energy in fusion-reactor designs relying on liquid lithium for cooling and breeding of tritium is the chemical energy represented by the lithium itself; and for such reactors, a lithium fire--whether initiated by internal or external events--may well represent the "maximum hypothetical accident". Lithium reacts vigorously and exothermically both with air and with water; like sodium, it also reacts with concrete (actually, with the water liberated from concrete by endothermic dehydration). Although the kinetics of these reactions are not well established experimentally, calculated maximum flame temperatures for both the lithium-air and lithium-concrete reactions are in the range of 2400 to 2500 K [VII-103]. These temperatures are below the melting points of refractory metals, such as niobium and TZM (titanium-zirconium-molybdenum), that might be used in fusion-reactor structures, but above the melting points of other potential structural materials, such as series SS 300. The refractory metals, although they would not melt, could be rather rapidly consumed at such temperatures by formation and volatilization of their oxides [VII-103]. The high temperatures and large energy releases potentially associated with lithium fires, therefore, pose the two-edged possibility of: *(a)* breaching multiple containment barriers between the reactor core and the public, and *(b)* augmenting the volatile tritium inventory that could escape through such breaks by converting activation products and toxic non-radioactive metals to volatile form.

The volatility of oxides of refractory metals can also pose problems under conditions much less extreme than those of a major lithium fire. Any malfunction that allows air to come into contact with structural refractory metals at their standard operating temperature of 800°C will lead to the oxidation of these materials, including their activated components, and their dispersal within the plant.

It must be emphasized, of course, that the existence of potential chains for the occurence of accidents--including some rather severe ones--does not necessarily mean that such accidents will occur frequently, or even at all. The relatively unfinished conceptual designs now available for fusion reactors already incorporate many features that are designed to minimize the probability of accidents, and to reduce the consequences if one occurs. These include double-walled piping, to reduce the chance of leaks and breaks; multiple separate lithium loops, to reduce the quantity that can be lost in any one break; stainless steel liners, to prevent spilled lithium from coming into contact with concrete; multiple, widely separated storage bunkers for tritium, to reduce the amount that could be lost in

a single event; and multiple containment barriers between stored energy sources and the outside world.

The value of increasingly detailed safety analyses at the early stages of design, of course, is that the designs can then evolve to cope with the most serious accident chains that are identified. The apparent flexibility of fusion in this respect is considerable, especially since passive systems seem capable of handling most if not all of the stored energy sources. The use of liquid lithium as a coolant and breeding medium for tritium--which produces the largest stored-energy threat in many fusion-reactor designs--has of course been questioned. Unfortunately, the alternatives are not without their own difficulties. Use of helium as a coolant in conjunction with breeding in lithium solids, for example, normally requires the use of beryllium as a neutron multiplier*; this material is extremely toxic, thus producing additional threats to occupational and public health, and it is scarce and expensive. Use of fluorine-lithium-beryllium molten salt (FLIBE) as an alternative coolant and breeding material, on the other hand, leads to materials compatibility problems, and to the production of a particularly dangerous form of hydrofluoric acid (wherein the hydrogen is tritium, and the fluorine is the two-hour half-life F18). Clearly, much additional work will be required to determine how much the *apparent* flexibility of fusion with respect to coolant and breeding media can *actually* be exploited to maximize safety.

2.2 Release of Non-radioactive but Toxic Substances

There are several elements in the current fusion reactor designs which would represent an inhalation hazard if they were to be spread throughout the atmosphere. The probability of such an event is extremely small, and the reader should realize that a quantitative assessment of this problem *cannot* be taken too seriously at this time. However, in order to obtain a qualitative picture of how toxic these elements are, we will use the same approach as is used in the subsection on radioactivity in order to calculate the required dilution factors. Again, the reader is cautioned that such numbers are only good in a relative sense.

A list of potentially toxic elements is given in Table VII-V, along with the representative inventory in selected reactors, the industrial threshold limit value (TLV), and the inventory divided by the TLV. The latter ratio represents the required dilution volume if *all* of the elements in the nuclear island were uniformly spread out into the atmosphere.

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* assuming that a breeding ratio greater than 1.1 is required, and approximately 10 to 20 per cent of the wall area is unavailable for breeding.

Table VII-V: Relative Chemical Inhalation Toxicity of Some Materials in Fusion Reactor Designs

Element	UWMAK-I	UWMAK-II	UWMAK-III	LASL-ANL	PPPL	Industrial Threshold Limit Value (mg/m^3)		Maximum Inventory (kg_{air}^3 / kW(e))
Be	None	0.3	None	0.01	0.4	0.002		0.2
B	1.1	2.5	0.47	-	-	15	(B_2O_3)	$1.6 \cdot 10^{-4}$
Cr	3.4	2.3	0.81	-	0.82	0.5	(solid compound)	$7 \cdot 10^{-3}$
Cu	5.1	6.1	0.65	2.3	1.6	0.1	(fume)	$6 \cdot 10^{-2}$
Pb	14	12	18	*	-	0.2		$7 \cdot 10^{-2}$
Li	1.2	0.2	0.2	0.4	0.3	2	(as LiOH)	$6 \cdot 10^{-4}$
						0.025	(as LiH)	$5 \cdot 10^{-2}$
Hg	0.002	None	None	-	-	0.05	(non-alkyl)	$4 \cdot 10^{-5}$
						0.01	(alkyl compounds)	$8 \cdot 10^{-6}$
Mo	0.39	0.24	0.96	None	None	10	(insoluble compound)	$1 \cdot 10^{-4}$
						5	(soluble compound)	$2 \cdot 10^{-4}$
Ni	2.6	1.7	0.49	None	0.81	1	(Ni or compound)	$3 \cdot 10^{-3}$

* Not specified.

The numbers in Table VII-V can be compared to those for radioactive species. The first point to note is that the dilution factor for the toxic substances is at least three orders of magnitude less than for the radioactive species. The second important point is that the potential dispersal of Be represents the most serious problem, while the hazards of Pb and Li cannot be neglected. The third point to note is that, once the hazard potential for the dispersal of Hg was recognized, reactor designs stopped using mercury diffusion pumps.

The problem with Be is normally connected to the use of solid breeders, and it is not inherent to fusion power if liquid Li is used as a breeder. The extreme toxicity and the potential for release due to some of the mechanisms outlined in subsection VII.2.1 suggest that one should carefully weigh the advantages and disadvantages of its use in high temperature environments.

3. WARFARE AND OTHER HYPOTHETICAL EVENTS

3.1 Thinking About the Unthinkable

However small the probability of a major accident may be or may seem to be, it is an important part of risk analysis to

identify the boundaries of the consequences of such events wherever possible. That is, it is *not* irrational (as has sometimes been asserted) to include in risk evaluation and comparison not only the expected consequences (i.e. the sum over all possible events of the probability of the event times the consequences of that event) but also the magnitude of the worst event, almost irrespective of its probability. The reasons why a rational analysis includes the latter are two-fold.

First, there are external events that can neither be readily incorporated into probabilistic analysis nor be defended against with absolute certainty. Sabotage and warfare come in mind. These may be capable of causing releases larger than any conceivable technological failure. Because of this possibility, any computation of the expected value of casualties from accidents is liable to be misleadingly incomplete.

Second, the social disruption and trauma caused by a single large and improbable event may be much larger than would be associated with a distributed set of events for which the expected number of casualties is the same. Consider the standard example of automobile fatalities in the USA of about 50,000 per year. A (hypothetical) type of event that killed one million Americans outright every 20 years would contribute exactly the same toll expected per year as the automobiles in a probabilistic analysis, but would be regarded by most people as a more horrifying possibility. Accordingly, they would be prepared to give up more in order to avoid entirely the risk of this event than they would be prepared to give up to avoid the risk of automobiles. This is in fact not irrational, and so it becomes necessary to face the task of calculating consequences of even highly improbable events.

3.2 Large Releases in Fission and Fusion Systems

In this section, we suppose for the sake of argument that, hypothetically, a mechanism has been identified whereby the principal barriers between the radioactive inventories in a fission or fusion system and the environment could be breached. The initiating event might be an aircraft impact or natural disaster more severe than the design capability of the facility, an act of war or sabotage, or a chain of events not anticipated by the designers. It is now important to emphasize the logical place of these considerations.

For fission, the distinction between realistic accidents and hypothetical accidents was made (see subsection VII.1.3). One hypothetical accident chain was evaluated, that was not known by a broader public to be inconceivable, and it was concluded that the fast breeder community considers this hypothetical accident chain as now proven to be inconceivable. Also, the chain of accidental events leading to the design basis

accident was evaluated and considered as ultimately conceivable but of very low probability, and design measures can make this very low probability to be below a preconceived level. In the case of the design basis accident the containment remains intact.

For fusion this ordering is necessarily less explicit as only conceptual designs exist. But it is equally reasonable to assume for fusion that for the design basis accident and beyond the containment would remain intact.

The viewpoint of this Section is to examine maximum hypothetical consequences of a major release of inventories; we intentionally do *not* go into an analysis of the underlying causes. We only have to specifiy that the mechanisms envisaged do break open the containment structures and thus severely damage the systems. The most obvious although not only case to refer to is warfare.

A second observation must be made. In Chapter V (Figure V-1), four levels of consideration are outlined that lead to injury of humans:

(1) total inventory of radioisotopes in the reactor;

(2) the maximum biological hazard potential in the reactor;

(3) the actualization of the biological hazard potential by movement of materials along pathways into and through the environment;

(4) injury to humans.

The first two levels are discussed in Chapter V. It is indeed important to distinguish between levels *(2)* and *(3)*. This can be most strikingly exemplified for actinides in the case of fission. At the end of Subsection VII.1.5, accidents beyond the design basis accident are considered ad hoc; it is assumed that up to 60 per cent of the core plutonium would be released in the form of fuel aerosols. Coagulations and sedimentations would lead to the settlement of most of the aerosols, and only 20 mg/m^3 can stay airborne for longer periods, and could be subject to leakages and releases by reventing. As explained there, the resulting irradiation dose at the fence of the reactor plant would be less than one rem total. Another example along these lines, but for fusion, involves the formation of volatile refractory metal oxides (i.e. MoO_3), which may plate out on cold surfaces before being widely dispersed. Thus the mechanisms of aerosols or vapors to stay airborne introduce a significant difference between levels *(2)* and *(3)*; this exemplifies the caution to be taken if only maximum biological hazard potentials (BHP) are being considered. This caution was advised throughout Chapter V.

The present example for differentiating between levels *(2)* and *(3)*, of course, must refer to a certain scenario for both the accident and the pathways for actualization of the biological hazard

potential into and through the environment. More generally, not only aerosol mechanisms have to be taken into account in other scenarios. This explains the difficulty of such investigations and related controversies. It is felt that comprehensive studies in depth of levels *(3)* and *(4)* are very much needed for the assessment of the nuclear fission and fusion options. We maintain that a comparison of fusion and fission must be made on level *(4)*.

Nevertheless, several studies of the consequences of very large, hypothetical fission-reactor accidents have been published (see, e.g. [VII-104 to VII-107]. These studies state or imply numbers of early fatalities, ranging from zero under scenarios of favorable conditions of meteorology and population distribution, to tens of thousands under scenarios with more adverse circumstances. "Early fatalities" are those occurring in the first month or two after the event as a direct result of acute radiation exposure. Other damages for scenarios considered in some of these studies include early non-fatal radiation injuries, delayed incidence of cancer and thyroid nodules, production of genetic defects, destruction of crops, and withdrawal of land from habitation and food production.

The most elaborate of the fission-accident studies published to date is the US Nuclear Regulatory Commisssion's *Reactor Safety Study* (popularly known as the *"Rasmussen Report"*, and referred to in the following as RSS), which considered light-water reactors of about 1000 MW(e) [VII-107]. The worst accident scenario considered there involved 3300 early fatalities, 45,000 early injuries, 45,000 cancer deaths in the forty years subsequent to the accident, 240,000 thyroid nodules, 28,000 genetic defects over the next several generations, about 14 billion dollars in property damage, and required the evacuation of 75,000 hectares of land. (The estimated probability of this event was 10^{-9} per reactor per year, excluding warfare and sabotage).

Several critiques of RSS have claimed that the "worst case" just described actually underestimates the possible consequences, supporting this view both with specific criticism of the RSS methodology and by noting the roughly tenfold higher estimates of early fatalities derived in earlier work [VII-108 to VII-110]. Nevertheless, it is instructive here to use some of the RSS methodology to arrive at *relative* hazard estimates for fission and fusion, wherein the principal assumptions are the same for both cases. Employing the RSS methodology and reducing the attention to relative hazard estimates is, for the moment, the best that can be done in analyzing levels *(3)* and *(4)* in view of a comparison of fission and fusion. More specifically, this analysis must involve the following steps:

- evaluation of the fraction of inventories released and the form of released materials;
- analysis of the dispersion of released materials in the environment;
- relation of environmental concentrations of radionuclides to absorbed doses of radiation in critical organs; and
- use of dose-response relations and population densities exposed by various pathways, to estimate expected casualties.

Significant uncertainties exist at each step.

Radioactive inventories were considered above in Chapter V, together with the measure of maximum biological hazard potential one obtains by dividing inventories (curies) by the maximum permissible concentrations (MPC) in air (Ci/m^3). Here we describe the remaining steps of an accident analysis--in the sequence just given--following the RSS methodology for computation of "critical doses" to persons in the vicinity of the accident. The critical dose with which we are concerned here is that which is relevant to determining early fatalities and, to some extent, early injuries. We have *not* carried out a comparison of late effects and property damage for hypothetical fission and fusion accidents, owing to the limited time available for this study; such a comparison should have high priority in future work.

The RSS group determined that the dose governing early fatalities was that which is delivered to the bone marrow; doses to the lungs and gastrointestinal tract might, in some cases, produce early injuries not predicted from the bone-marrow dose, but the latter seems to be the best single predictor of early effects. The RSS defined the "critical dose" to bone marrow, for the purpose of predicting these early effects, to be equal to the sum of (RSS Appendix VI, Chaper 9):

(a) external (γ-ray) dose from passing cloud;

(b) external (γ-ray) dose from emitters deposited on ground;

(c) internal dose received during the first seven days from radionuclides inhaled during cloud passage; and

(d) one half of the internal dose received from day 8 through day 30 from radionuclides inhaled during cloud passage.

The external dose from contaminated ground was truncated in the RSS after four hours for individuals within 25 miles (40 km) of the reactor, owing to the presumed success of evacuation procedures. Beyond 25 miles the ground dose was truncated at seven days for purposes of this "critical dose" calculation.

Calculations of the critical dose just described for large releases of mixed fission products are especially difficult because different combinations of isotopes are dominant in determining the different components of the dose: iodine, and krypton dominate the external cloud dose; tellurium, strontium, barium, and cesium govern the inhaled dose. The RSS calculated accident consequences for a variety of different accident modes, labeled PWR-1, PWR-2, ...,BWR-1, BWR-2, ..., which are characterized by various combinations of release fractions for various classes of isotopes. The only calculation presented in RSS in a way that quantitatively disaggregates the contributions of different isotopes to different dose pathways, however, corresponds to none of the tabulated release categories. (The relative release fractions of various classes of isotopes for this sample calculation--described in RSS Appendix VI, Chapter 13 as a "large, cold, ground-level release"--are not stated there explicitly, but they can be deduced from details of the calculation that are provided.) If the deduced relative release fractions for the RSS sample calculation are scaled to produce an absolute release fraction of 0.4 for iodine, one obtains an accident significantly less severe than PWR-1 or BWR-1, and somewhat more severe than PWR-3 or BWR-3 (see Table VII-VI).

This sample consequence calculation, in the following refered to as RSS-VI-13, is the only one in RSS that provides enough detail to permit a directly comparable calculation of fusion accident consequences; so we use it here. Two points should be emphasized, however.

First, RSS-VI-13 is by no means the most severe release imaginable in a light water reactor; PWR-1, PWR-2, BWR-1, and BWR-2 are all more severe, as Table VII-VI indicates, and some critics of the RSS have argued that even PWR-1 and BWR-1 are not the most severe releases imaginable [VII-110]. Second, the inventories and release fractions in the worst LMFBR accident will no doubt differ somewhat from those used in RSS for LWR accidents. The LWR inventories used in RSS for the isotopes that dominate the "critical dose" calculation are summarized in Table VII-VII. These may be compared with the LMFBR inventories in Chapter V above. Release fractions for some isotopes may differ more substantially. A difference in inventories between LMFBR and LWR appears if the LWR is assumed to be without Pu recycling. In that case the LMFBR Pu inventory is three times larger. If the LWR with Pu recycling is considered, the difference is less than a factor of two (see Table VII-VIII). The RSS study assumes a release fraction for the actinides of 0.5 per cent. This confirms the difference mentioned above between the maximum biological hazard potential in the reactor and that actually going into the environment (difference between levels *(2)* and *(3)*). With such a release rate the critical dose for early fatalities will not increase appreciably, and we assume the same for the LMFBR. (But even with the same release fraction as for LWRs, the increased actinide inventory in

Table VII-VI: Release Fractions for Various Classes of Isotopes in Different Severe Hypothetical Accidents in Light Water Reactors, as Treated in the Reactor Safety Study[a]

Accident	Iodine	Tellurium-Antimony	Cesium-Rubidium	Barium-Strontium	Ruthenium[b]
Sample calculation in Appendix VI Chapter 13	0.4[c]	0.4	0.2	0.05	0.03
PWR-1 [d]	0.7	0.4	0.4	0.05	0.4
PWR-2 [d]	0.7	0.3	0.5	0.06	0.02
PWR-3 [d]	0.2	0.3	0.2	0.02	0.03
BWR-1 [d]	0.4	0.7	0.4	0.05	0.5
BWR-2 [d]	0.9	0.3	0.5	0.10	0.03
BWR-3 [d]	0.1	0.3	0.1	0.01	0.02

[a] Rasmussen Report or RSS [VII-107];

[b] Includes Ru, Rh, Co, Mo, Tc;

[c] Relative release fractions were derived from RSS, Appendix VI, Figures VI-13-1 and VI-13-2, and Table VI-D-2. These were then scaled to give an absolute release fraction for iodine of 0.4;

[d] Taken directly from RSS, Appendix VI, Table VI-2-1.

LMFBRs would greatly affect the number of latent cancers produced by a large accident. For this reason and others, focussing only on early deaths, as we do here, may give a distorted comparison of the fission and fusion cases. We emphasize again, therefore, the desirability of a more comprehensive calculation.)

The atmospheric dispersion model used in the RSS as the next step after the determination of release fractions is a version of the well-known Gaussian plume model. The integrated ground-level cloud exposure experienced in the path of the release (of assumed half-hour duration) is given in the simplest case (cold release at ground level) by (RSS, Appendix VI, page 4-1):

$$X(x) = 2Q/3\sqrt{2\pi} \cdot U \cdot \sigma_y(x) \cdot \sigma_z(x) \qquad (1)$$

Table VII-VII: Inventories of Dominant Fission Products in a 1000 MW(e) PWR at Accident Initiation*

Nuclide	Inventory (MCi)	Specific Inventory (Ci/kW(th))
Krypton 88	68	23
Strontium 89	94	31
Tellurium 132	120	40
Antimony 129	33	11
Iodine 131	85	28
Iodine 132	120	40
Iodine 133	170	57
Iodine 135	150	50
Cesium 134	7.5	2.5
Cesium 137	4.7	1.6
Barium 140	160	53
Lanthanum 140	160	53

* RSS, Appendix VI, Table VI-3-1 [VII-107].

Table VII-VIII: Radioactive Inventories of Fission Reactors after Shut-down (Ci/kW(th))

		Time After Shut-down 10^3sec	10^7sec	10^8sec	10^{10} sec
Pu241	LWR	3.9	3.9	3.3	-
	LMFBR	7.3	7.3	6.2	-
$\sum_{239-242}$ Pu	LWR	3.9	3.9	3.4	0.03
	LMFBR	7.5	7.5	6.4	0.17

Note: LWR means LWR with a U:Pu power ratio of 0.7:0.3, and Pu239:Pu240:Pu241:Pu242 = 0.59:0.26:0.12:0.03, and a rating of 1 MW(th)/kg fissile material;

LMFBR means LMFBR with Pu as fuel in natural uranium and Pu239:Pu240:Pu241:Pu242 = 0.69:0.25:0.04:0.02, and a rating of 1 MW(th)/kg fissile material.

where

X is in $Ci \cdot sec/m^3$;

x is downwind distance from the accident (m);

Q is the number of Ci released;

U is mean wind speed (m/sec);

$\sigma_y(x)$ is the cross-wind dispersion coefficient; and

$\sigma_z(x)$ is the vertical dispersion coefficient.

The RSS employs a numerical fit to the empirically determined Pasquill-Gifford values of the cross-wind and vertical dispersion coefficients for different atmospheric stability conditions. (Several approaches to this aspect of the problem are described in RSS, Appendix VI, pages A-4 to A-6; a clear statement on which one was actually used appears in Appendix VI, page 4-2.) The vertical coefficient is not allowed to exceed 0.8 of the atmospheric mixing height (typically 250 to 3000 m, Appendix VI, page 5-6). The exposure given by equation (1) is assumed to be uniform across a plume width of $3\sigma_y$, at the boundaries of which it drops to zero.

For all but noble gases, equation (1) is modified to account for the depletion of the cloud by dry fall-out by replacing Q with $Q \cdot f(x)$, where the depletion correction f(x) is approximated for ground-level releases by

$$f(x) = \exp[-v_d \cdot x/U \cdot (\pi/2)^{1/2} \cdot \sigma_z] \quad . \qquad (2)$$

The dry deposition velocity v_d is an empirically determined parameter; it is taken by RSS to be 10^{-2} m/sec for all isotopes (other than noble gases, for which $v_d = 0$). RSS gives an uncertainty of a factor of ten larger or smaller on this parameter (RSS, Appendix VI, page B-9). Most of the consequence calculations in RSS also correct equation (1) for bouyant plume rise owing to the thermal energy content of the release (RSS, Appendix VI, page A-8), and for the effect of the reactor buildings on plume aerodynamics at short distances (page A-12). In the present work, we have included dry deposition but neglected thermal plume rise and building-wake effects. The latter should be added when more thorough comparisons are made.

Use of equations (1) and (2) for a given atmospheric stability condition yields X values for each released radionuclide as a function of x. Number of curies inhaled by an individual within the band of width of $3\sigma_y$ spanned by the plume is then obtained as the product of X with the breathing rate, taken by RSS to be $B = 2.66 \cdot 10^{-4}$ m^3/sec for adults. Curies of a given isotope inhaled can then be converted into absorbed 7-day and 30-day bone-marrow doses by multiplying them by the dose-conversion factors D (in rem/Ci), in RSS Appendix VI, Table VI-D-2 (page D-8).

As pointed out in RSS, Appendix VI, Chapter 13, the *relative* contributions of all isotopes and pathways (inhalation, cloud γ dose, ground γ dose) to the critical dose do not vary with atmospheric conditions or distance for times that are short enough so that differences in half-life do not matter, as long as all isotopes have the same deposition velocity (v_d). Thus, if the relative contributions to the critical dose are known at one point, it suffices to follow a single isotope and pathway (we chose inhaled Te132), and scale the others to it, using the known fixed ratios. The assumptions are not met perfectly: the noble gas Kr88 is important in governing the external-cloud γ dose, and it has a deposition velocity (zero) that is different from those of the other important isotopes; for low wind speed and long distances downwind, differential depletion by radioactive decay of a few isotopes is of some significance. But the errors should be modest in the first few tens of kilometers, where the "critical dose" to bone marrow is high enough to produce early fatalities or early injuries.

For sample release RSS-VI-13, examination of Figure VI-13-1 in RSS Appendix VI shows that the 30-day bone marrow dose due to inhaled Te132 is equal to 10.4 per cent of the critical dose to bone marrow as defined above. (Total cloud dose = 484 units, 7-day dose from inhalation = 245 units, half of 8-day to 30-day dose from inhalation = 40 units, 4-hour ground dose = 383 units; sum = critical dose = 1152 units; 30-day dose from Te132 inhalation = 120 units = 10.4 per cent of 1152.) Thus, the critical dose at all distances and atmospheric conditions of interest can be obtained for release RSS-VI-13 by computing X from equations (1) and (2) for a release of $Q = 0.4 \cdot 120 \cdot 10^6$ Ci of Te132, obtaining the 30-day inhalation dose as $X \cdot B \cdot D$ (with D = 1000 rem/Ci from RSS Table VI-D-2), divided by 0.104. For our reference conditions we take atmospheric stability condition Pasquill F (very stable), and mean wind speed U of 1 m/sec. There follow the values of X, 30-day inhalation dose for Te132, and the "critical dose" to bone marrow given in Table VII-IX.

If we assume that the "critical dose" to bone marrow in the fusion case is dominated by the release of tritium, the corresponding calculation is relatively straightforward. Tritium emits only a soft β particle (average energy 5.7 keV, maximum 18 keV); since this radiation cannot penetrate the skin, the only dose to bone marrow is from tritium taken in as tritiated

Table VII-IX: Critical Doses to Bone Marrow from Fission Product Release RSS-VI-13 [VII-107]*

x (km)	X^{Te132} (Ci·sec/m³)	30-day Inhalation Dose from Te132 (rem)	Critical Dose to Bone Marrow (rem)
0.1	$1.18 \cdot 10^{6}$	$3.13 \cdot 10^{5}$	$3.01 \cdot 10^{6}$
0.2	$3.03 \cdot 10^{5}$	$8.06 \cdot 10^{4}$	$7.75 \cdot 10^{5}$
0.5	$5.04 \cdot 10^{4}$	$1.34 \cdot 10^{4}$	$1.29 \cdot 10^{5}$
1.0	$1.47 \cdot 10^{3}$	$3.91 \cdot 10^{3}$	$3.76 \cdot 10^{4}$
2.0	$4.06 \cdot 10^{3}$	$1.08 \cdot 10^{3}$	$1.04 \cdot 10^{4}$
5.0	$7.27 \cdot 10^{2}$	$1.93 \cdot 10^{2}$	$1.86 \cdot 10^{3}$
10.0	$1.55 \cdot 10^{2}$	$4.12 \cdot 10^{1}$	$3.96 \cdot 10^{2}$
20.0	$2.71 \cdot 10^{1}$	$7.21 \cdot 10^{0}$	$6.93 \cdot 10^{1}$
50.0	$7.80 \cdot 10^{-1}$	$2.07 \cdot 10^{-1}$	$1.99 \cdot 10^{0}$
100.0	$1.07 \cdot 10^{-2}$	$2.85 \cdot 10^{-3}$	$2.74 \cdot 10^{-2}$

* Atmospheric stability Pasquill F, mean wind 1 m/sec, $v_d = 10^{-2}$ m/sec.

water and incorporated into body water. Taking the biological half-life of tritium in humans to be 12 days [VII-111], the relative biological effectiveness of tritium's β radiation as 1.0 [VII-112], and assuming uniform irradiation of body tissues having a mass of 70 kg, the dose commitment is readily calculated as 77 rem per Ci of tritium taken in as tritiated water. (Tritium as hydrogen gas is by orders of magnitude less dangerous, as almost none is absorbed when it is inhaled.) The "critical dose" by the RSS formula (7-day dose plus half of 8-day to 30-day dose) is 45 rem per Ci of tritiated water taken in.

We have used the identical plume formulas and deposition velocity as for the fission case to calculate the X values for a release of 10^8 Ci of tritiated water in a fusion accident. This corresponds to 0.4 of the tritium inventory of a 2500 MW(th) fusion reactor at 10 kg tritium per 1000 MW(th) (see Chapter V above). Whether so large a release of tritium as the oxide in a short time period must really be considered is questionable, but assuming this fraction gives a somewhat satisfying comparison with the fission case, where release fractions for iodine and tellurium isotopes have also been taken to be 0.4 In converting X to the critical dose from tritium we account for skin absorption as well as for inhalation; this adds

$2.3 \cdot 10^{-4}$ m^3/sec to the effective breathing rate, giving $B = 5 \cdot 10^{-4}$ m^3/sec for this case. The pathways through food and water are not considered in this "critical dose" calculation for the same reason why they were excluded in the corresponding calculation in RSS: individuals in the narrow path of the plume would be most unlikely to find food and water whose exposure history was nearly as unfortunate as their own. In computing the lower doses received by much larger populations at longer distances, these pathways of course would have to be considered in both the fission and fusion cases.

The critical doses to bone marrow for the tritium case are given in Table VII-X. The land areas corresponding to each dose are given in the last column of that table; these have been obtained from the assumption (verified graphically) that the computed plume is approximately triangular, hence for any downwind distance it has an area of $1.5\ x \cdot \sigma_y(x)$. The same areas correspond to the same downwind distances in the fission case, since the plume calculations are identical.

Critical dose to bone marrow versus area is plotted for the fission and fusion cases in Figure VII-11, based on the data in Tables VII-VII and VII-X. RSS estimates that the $LD_{50/60}$

Table VII-X: Critical Doses to Bone Marrow from Release of 10^8 Curies of HTO*

x (km)	X^{HTO} (Ci·sec/m^3)	Critical Dose to Bone Marrow (rem)	Area Receiving This Dose or Greater (km^2)
0.1	$2.45 \cdot 10^6$	$5.51 \cdot 10^4$	0.0006
0.2	$6.32 \cdot 10^5$	$1.42 \cdot 10^4$	0.0024
0.5	$1.05 \cdot 10^5$	$2.63 \cdot 10^3$	0.015
1.0	$3.07 \cdot 10^4$	$6.91 \cdot 10^2$	0.054
2.0	$8.46 \cdot 10^3$	$1.90 \cdot 10^2$	0.21
5.0	$1.51 \cdot 10^3$	$3.40 \cdot 10^1$	1.2
10.0	$3.22 \cdot 10^2$	$7.25 \cdot 10^0$	4.5
20.0	$5.64 \cdot 10^1$	$1.27 \cdot 10^0$	16.5
50.0	$1.62 \cdot 10^0$	$3.65 \cdot 10^{-2}$	105.
100.0	$2.42 \cdot 10^{-2}$	$5.45 \cdot 10^{-4}$	360.

* Atmospheric stability Pasquill F, mean wind speed 1 m/sec, deposition velocity 10^{-2} m/sec.

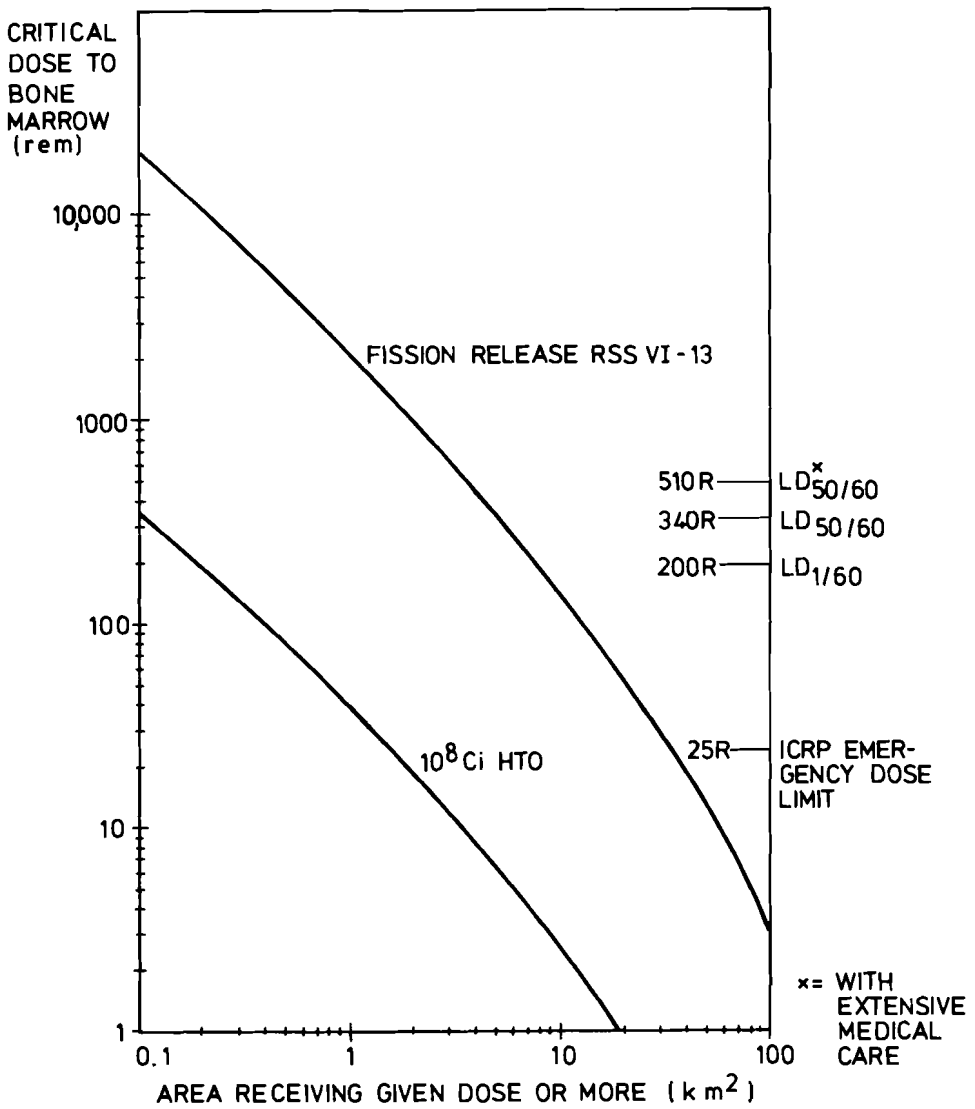

Figure VII-11: Critical Dose versus Area for Severe Releases in Fission and Fusion Systems

(magnitude of critical dose to bone marrow at which, with only minimal medical treatment, half the victims will die within 60 days) is 340 rad (RSS Appendix VI, page 9-3). (Rad is the same as rem if the relative biological effectiveness (RBE) is 1.0, as is the case for the radiation of importance in both cases here.) The $LD_{50/60}$ with intensive supportive medical treatment was estimated by RSS to be 510 rad. The dose at which one per cent would die within 60 days with minimal medical treatment was about 200 rad.

From the figure one finds that the fission release delivers the $LD_{50/60}$ over 5 km^2, and the $LD_{1/60}$ over about 8 km^2, areas that are 40 to 50 times larger than those in the fusion case. By choosing a number for population density, one can transform these figures into estimates for actual numbers of early fatalities; at 300 people/km^2, the highest density considered in RSS but by no means the highest possible, the number will be some tens of people in the fusion case, and on the order of one thousand in the fission case.

It is important to emphasize that the relative magnitudes are more significant here than the absolute numbers. We have analyzed very severe releases but not necessarily the most severe possible, and the methodology of RSS may not yield an accurate consequence estimate in absolute terms. But by applying the *same* methodology to both fission and fusion, one obtains a first instructive comparison of relative consequences. Within all the constraining assumptions explained above, we now conclude that, for releases of large parts of the inventories of fusion or fission reactors by catastrophic events such as warfare, the fusion reactor is better off by a factor of 40 to 50 if compared to a fission reactor.

One weakness in our comparison is the failure to consider doses organ by organ (lung, gastrointestinal tract, etc.). Assuming RSS is correct in saying that the bone-marrow dose dominates fatalities, it is nevertheless true that the high doses delivered to individual organs by certain isotopes (ruthenium to the lung, strontium to the mineral bone, tellurium to the gastrointestinal tract) will cause early injuries and delayed cancers out of proportion to the bone-marrow dose. Since tritium is uniformly distributed in the body and causes no such disproportionate effects, leaving out the organ-by-organ dosimetry tends to understate fusion's advantage. Leaving out long-term doses from surface contamination and pathways to man through food chains may also understate fusion's advantage, because tritiated water is dispersed quickly and does not reconcentrate. The effect of this difference on late effects might not be very large if the linear hypothesis holds, because then a small dose to a large number of people (e.g., delivered by dispersed tritium) will cause the same number of adverse effects as the same number of person-rem delivered as larger doses to fewer people (e.g., by concentrated cesium 137); this aspect requires closer investigation.

Another major weakness is that the size of the considered releases is less well established for fusion than for fission. Since fusion-reactor designs are still only conceptual, one knows accurately neither the total tritium inventory nor the fraction that could really be released as HTO by the largest disasters. They depend on the actual design. The fission-product inventory in a fission reactor of specified size, by contrast, is fixed by laws of physics; one cannot reduce it by being clever. Moreover, releases more severe than that of RSS VI-13 are certainly conceivable--as noted earlier, the events PWR-1 and BWR-1 considered in RSS are significantly worse.

It may be argued, on the other hand, that our comparison overstates fusion's advantage by leaving out the potential release of structural activation products in the fusion accident. This may be so. According to Chapter V.3 above, the biological hazard potential of the best designs from the activation standpoint is in the vicinity of 40 km^3 of air per kW(th); 10^8 Ci of tritium per 1000 MW(th) amounts to a BHP of only 0.5 km^3 of air per kW(th). Thus, if more than one per cent of the activation products could be released, they might be a bigger hazard than the tritium. Some of the important activation products are refractory metals, hence presumably hard to release; but even some of these can form oxides that are volatile at accident temperatures. The question of release fractions for fusion-reactor activation products requires much closer attention, and this is consistent with what was observed earlier that, in any event, release mechanisms and pathways into and through the environment must be more fully investigated. Only then can a more thorough comparison between fusion and fission be made.

4. CONCLUSIONS

Early concerns about the safety of LMFBR focused on control characteristics and the possibility of core recompaction in accidents that begin with sodium boiling and local fuel melting. These concerns were accentuated by the emphasis on compact cores and metallic fuel elements in breeder designs of the 1950s and early 1960s. For the case of the large cores and mixed-oxide fuels typical of all prototype and commercial LMFBR designs in the 1970s, it is now known that in their crucial respects the control characteristics are substantially similar to those of the LWR. Moreover, a large and growing body of theoretical and experimental evidence supports the view that the propagation of local fuel failures in a way that leads to recompaction in the large-core, mixed-oxide fueled LMFBR would require combinations of events and degrees of spatial and temporal coherence that are not physically realistic.

The large LMFBR prototypes that are in operation in France and in advanced stages of construction in the FRG have undergone licensing reviews as stringent with respect to safety as the

ones that are applied to the LWR. The design basis accidents (DBA) for these large LMFBR encompass the possibility of failure of both independent shut-down systems, following a hypothetical large insertion of reactivity or coast-down of the main sodium pumps. The calculated consequences of melting and core disassembly in these maximum hypothetical accidents define the design characteristics of the containment systems required for licensing (strength of reactor vessel and primary piping; strength and leak rates of surrounding double steel and concrete containment structures). In addition to the pressure loads during a DBA, large commercial LMFBR would also have to cope after a DBA with long-term cooling of large masses of molten and dispersed fuel. While this capability appears to be at hand for the 300 MW(e) class LMFBR, additional development work is needed for larger LMFBR power stations. Meeting these design requirements, which as the French and German experience indicates can be done with reasonable technical effort, can restrict radiation doses to 1 rem or less at the plant boundary in the event the DBA occurs. The overall conclusion is that the LMFBR can meet the same predetermined safety standards as are applied to other fission reactors. This will also hold for the LMFBR fuel cycle (fabrication and reprocessing plants)[VII-113].

In the case of fusion, reactor safety analysis is necessarily much more primitive because the technology cannot yet be described in detail. Examination of stored energies and potential pathways for energy release in conceptual controlled thermonuclear reactor (CTR) designs indicates that sudden failures of the magnet support and vacuum systems could produce enough mechanical energy to severely damage the reactor. Loss of coolant or coolant flow coupled with failure to shut down the fusion reactor could cause local interior structural damage. The characteristics of fusion plasmas and the very small amount of fuel present in the reaction chamber at any time mean that reactivity accidents will not be an important concern. Decay-heat due to neutron activation of structural materials is small enough in most designs to be substantially easier to handle than in fission reactors. For CTR designs where liquid lithium serves as breeding medium and coolant, the very large chemical energy stored in this coolant and the high flame temperature of the lithium-air and lithium-water reactions (somewhat worse in both respects than the sodium in a comparable LMFBR) probably represent fusion's most important vulnerability to accidents capable of releasing sizable quantities of radioactivity. Both LMFBRs and liquid-lithium cooled CTRs require careful design of steam generators to handle safely the possibility of leaks that bring water into contact with liquid metal.

Many of the possible accident pathways for CTR can be minimized in importance by intelligent design, which includes the apparent possibility of tritium breeding in ceramic lithium compounds, and cooling with pressurized helium instead of liquid lithium. Such an approach may also be able to reduce the in-

ventory of blanket tritium that could be released in an accident, but other sources of tritium in proximity to the blanket (i.e. vacuum pumps, diverter collector plates, etc.) will not be affected by the change to solid breeders. Enthusiasm about the potential flexibility in CTR designs must be tempered with the recognition that there may be important trade-offs--for example, the probable need to use toxic and relatively scarce beryllium for neutron multiplication if solid breeders are employed in realistic blanket designs. Designers of fusion systems can anticipate that the approaches they devise to control energy release from magnets, vacuum systems, coolant, and so on will doubtless be subjected to much the same critical scrutiny and demand for high reliability experienced now in fission-reactor licensing proceedings.

It is possible to make a very crude comparison of fission and fusion reactors with respect to the consequences of events worse than the design basis accidents--resulting, for example, from acts of war, sabotage, or hypothetical events exceeding the design capabilities of the safety systems. Applying the consequence model of the Reactor Safety Study (Rasmussen Report) of the US Nuclear Regulatory Commission shows that hypothetical release of a substantial fraction of the fission products and 0.5 per cent of the actinides in an LWR (release PWR-1 in the Reactor Safety Study) would produce roughly 100 times more early deaths under adverse meteorological conditions than a release of 10^8 Ci of tritium oxide from a CTR under the same conditions. Much in need of further investigation is what fraction of the activation products in a CTR and of the actinides in an LMFBR could be released in such hypothetical events, as these could significantly affect the calculated outcomes. Comparative examination of delayed as opposed to early casualties is also needed.

In the CTR most of the fuel cycle is within the reactor containment structure in the form of the tritium cycling systems. For the LMFBR there must exist in addition fuel reprocessing and fuel fabrication plants, and the potential for accidents on the way to or at these facilities needs careful examination. Such analysis is not yet nearly as refined as that of the LMFBR itself. Both LMFBR and CTR will require some form of radioactive waste management facilities, for which accident analysis will also have to be done.

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VIII. SAFEGUARDS

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1. INTRODUCTION

Of interest here are the possible connections between the use of nuclear reactors and associated facilities on the one hand, and the production of nuclear weapons (explosive or radiological) on the other. To address this complicated issue systematically it is necessary to establish clearly a number of distinctions. Specifically, one should distinguish between the spread of weapons-related knowledge and the spread of weapons-related materials; between the threat of weapons production by governments and by subnational groups; between detection of diversion of material or knowledge and physical protection against diversion; between explosive and radiological weapons; and between the threat of diversion itself and the threat (or costs) that might be created by safeguards measures. Here we wish not only to compare fission and fusion in these respects but also to determine whether significant differences arise from the approach taken *within* fission or fusion (that is, for example, LMFBR versus LWR in fission, magnetic confinement versus laser-pellet systems in fusion). All our discussions here are based entirely on the unclassified literature.

2. NUCLEAR EXPLOSIVES

It is widely reported in the unclassified literature that the technical knowledge required to design a *fission* bomb is by now widespread, which means that many national governments already have this knowledge and that many more could acquire it if they chose to do so. Similarly, it is generally agreed that a modest degree of industrial sophistication suffices for a nation actually to construct such a bomb (to be distinguished from merely designing one) if it possesses the necessary fissile materials [VIII-1, VIII-2]. It is, therefore, fair to say that the international spread of fission weapons (proliferation) at the present time is limited by some combination of national intentions (nations choosing not to make weapons), reluctance to commit the necessary resources of manpower and money, and lack of fissile materials (which until recently have been producible in quantity only by the most advanced industrial nations).

Thus there is no threat that the spread of fission reactors will spread dangerous knowledge, for that spread has already taken place. There is a threat, however, that the spread of reactors will give access to fissile materials to nations that do not now have such access; some may choose to use this opportunity to construct bombs. (Moreover, nations that do not do so at once may be tempted to do so under different political circumstances in the future). This international proliferation of nuclear weapons by way of commercial nuclear power is the phenomenon which the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) and associated International Atomic Energy Agency (IAEA) safeguards are designed to prevent (see below). The main controversies on this specific aspect of the diversion problem can be summarized by the following questions: *(1)* How effective are the NPT/IAEA safeguards at detecting diversion? *(2)* Would nations that wish to have nuclear weapons be deterred from using reactor-produced materials for this purpose even if detection and international censure were certain? *(3)* Are other means of obtaining fissile materials (e.g. enrichment of uranium with centrifuges, nozzles, or lasers) becoming so widely available that the spread of nuclear reactors is unimportant by comparison?

The problem of production of fission bombs by subnational groups--organized crime, terrorists, etc.--is even more controversial. At issue here is: *(1)* whether such groups could design and construct a bomb that would work, given the materials, and *(2)* whether existing and foreseeable physical protection measures could prevent determined groups from obtaining these materials. On the first point, T.B. Taylor has argued forcefully that a technically sophisticated group with a few thousand dollars worth of equipment could succeed [VIII-3]. The US Nuclear Regulatory Commission argues, on the other hand, that there are many more wrong ways of making a fission bomb than right ways, and that the most likely outcome is that the individuals trying it would kill themselves but no one else [VIII-4]. No meaningful assessment of the *probability* of a successful bomb (given the materials) has come out of this debate, and none is likely to.

Not in dispute among informed observers, however, are several relevant technical facts. The easiest material for relatively unsophisticated bomb-makers to use is highly enriched uranium. Plutonium is more difficult, and plutonium with the high plutonium 240 content associated with LWR fuel cycles ("reactor-grade" plutonium) is more difficult still. Use of the oxides is more difficult than use of the metals. With increasing sophistication on the part of the bomb-makers, however, all of these materials can be used to make workable nuclear explosives [VIII-3, VIII-5], although the common chemical form, plutonium nitrate, cannot be used. Moreover, in some of the cases that matter, the explosive yield will be quite unpredictable, but this is probably not so important for the purposes of terrorists, blackmailers, etc. Thus, one must conclude that a safeguards problem already exists for light-water reactors; indeed, it is a feature of nuclear fission power in general, not one specific

to the fast breeder reactor.

Concerning adequacy of physical protection measures, there is general agreement that the measures in effect in the early 1970s in the United States (at least) would not have been adequate to prevent a determined group from seizing bomb quantities of fissile materials [VIII-3, VIII-6]. Whether the much more stringent and elaborate measures now coming into effect will be adequate is controversial; for obvious reasons, the full details of such measures have not been published in the open literature and are not likely to be. The public's dilemma is thus two-fold: How can it be confident in a system that cannot be described? If there is a finite residual probability of a successful nuclear explosion by criminal elements, do the benefits of nuclear power justify this *kind* of risk? These are of course socio-political questions and not strictly technical ones. (Clearly, the magnitude of the risk can be influenced by technical means; see below.)

The link between *fusion* and bombs is different. Fusion bombs are unquestionably harder to design and construct than are fission bombs, and the spread of fusion bombs has not been limited at all by lack of fusion fuels but rather by lack of knowledge and technical sophistication. As far as can be determined from the unclassified literature, all successful fusion bombs have required the use of a fission bomb for a trigger [VIII-7]. It is fair to assume that the design of a "pure fusion" bomb is a much more difficult task than the already hard one of making a fusion bomb with a fission trigger. In any event, any group or nation sophisticated enough to design and fabricate a workable fusion explosive, with or without a fission trigger, will almost certainly be sophisticated enough to produce any fusion fuels it needs without having access to a fusion reactor. (It is at least conceivable in principle, however, that access to tritium from a fusion reactor might simplify the task enough to make its diversion or theft worthwhile.)

One possible connection between fusion reactors and fissile materials must be mentioned. Once a working fusion reactor is in hand, it probably will not be difficult to use it for the production of fissile plutonium by bombarding natural uranium with the energetic fusion neutrons (see also the discussion of fission-fusion hybrids, Appendix B). In the international safeguards debate this problem has been described as "the diversion of neutrons". This could only be done with the knowledge and full cooperation of the fusion reactor's operators, of course, and it would be impossible to conceal if the reactor were subject to any sort of national or international inspection.

With respect to the spread of *knowledge* related to fusion bombs, a distinction must be made between the magnetic-confinement approach to controlled fusion and the inertial confinement approach. It seems clear on basic physical grounds that magnetic-confinement

approaches are not relevant to the problem of designing a fusion bomb, and thus that the spread of magnetic-confinement fusion technology poses no realistic threat of spreading dangerous knowledge. Magnetic-confinement fusion research was declassified by an international agreement in 1958 and has remained unclassified in all countries. By contrast, one may surmise, from the fact that much inertial confinement research remains classified in the major countries pursuing it, that it has some relevance to the design of fusion weapons. Thus it is also conceivable that the spread of a successful inertial confinement reactor technology could, by means of diffusion of specific kinds of technical knowledge, facilitate the international spread of thermonuclear weapons capability.

3. RADIOLOGICAL WEAPONS

For purposes of blackmail and terrorism, subnational groups may find it easier to use nuclear materials as radiological poisons than to design and manufacture nuclear explosives. Of greatest concern are plutonium in the case of fission systems and tritium in the case of fusion. (Other radioactive materials, such as fission products and activation products, could also be used, but these would be much more difficult for subnational groups to transport and to handle.)

Scenarios for how such radiological poisons could be dispersed and estimates of the consequences have been developed elsewhere [VIII-7, VIII-8]. It suffices to say here that the readily attainable consequences seem less than the consequences of a successful nuclear explosion in a city, but large enough to necessitate very stringent safeguards against diversion even of quantities of material much too small to make a bomb.

It is perhaps important to ask whether other, non-nuclear methods of poisoning people are so easy to perform that radiological terrorism adds little to society's overall vulnerability, but we do not propose to try to resolve this question here. We restrict ourselves instead to a brief comparison of the characteristics of plutonium and tritium as radiological poisons.

Some of the data required to assess the relative magnitudes of these threats are summarized in Table VII-I. The maximum permissible concentrations (MPC) for plutonium isotopes are much lower than those of tritium (measured as Ci/m^3), but the specific activity of tritium (Ci/gram) is much higher than for plutonium [VIII-9]. The result is that the biological hazard potentials (BHP) associated with the inventory per GW(e) or flow per GW(e)·yr of tritium and reactor-grade plutonium work out as follows: tritium is less dangerous than plutonium by two to five orders of magnitude with respect to contamination of air, but the two are approximately equal with respect to contamination of water. Some degree of consensus appears to be emerging

Table VIII-I: Radiological Hazards of Plutonium and Tritium (quantities normalized where appropriate to 1 GW(e) of capacity)

		Plutonium	Tritium
Inventory Outside Blanket	kg	900	25
Annual Flow Outside Reactor	kg	1600	32[a]
MPC_{air}	Ci/km^3		
Insoluble Pu239, HT or T_2 gas		0.001	40,000
Soluble Pu239, HTO vapor		0.00006	200
BHP	km^3 of air per gram of		
Pure Pu239, elemental T		63 to 1000	0.25
Reactor Pu[b], T in HTO		300 to 5000	50
BHP/GW(e)·yr[c]	10^6 km^3 of air per year		
Best case[d]		450	0.008
Worst case[e]		7500	1.6
BHP/GW(e)[f]	10^6 km^3 of air		
Best case[d]		270	0.006
Worst case[e]		4500	1.25
MPC_{water}	Ci/km^3 for soluble forms	5000	3,000,000
BHP	m^3 of water per gram of		
Pure Pu239 in soluble compound		12,500	
Reactor Pu in soluble compound		62,500	
Pure T in HTO			3,300.000
BHP/GW(e)·yr[c]	km^3 of water per year	94	110
BHP/GW(e)[f]	km^3 of water	56	83

(a) At breeding ratio = 1.25;
(b) Contains Pu238, Pu239, Pu240, Pu241, Pu242;
(c) Based on flow outside reactor;
(d) Reactor-grade Pu dispersed in insoluble form, tritium dispersed as T gas;
(e) Reactor-grade Pu dispersed in soluble form, tritium dispersed as HTO vapor;
(f) Based on inventories outside blankets.

that the MPCs for plutonium are too high by a factor of five to ten (in a relative sense, the comparison being in this case with radium) [VIII-10, VIII-11], and there is some reason to think the MPCs for tritium may be too low by factors of two to four (see discussion above). Incorporation of such correction factors, if they are validated, would change the foregoing comparison to make tritium less dangerous than plutonium by one order of magnitude with respect to contamination of water, and to give tritium an advantage of three to six orders of magnitude with respect to contamination of air. In terms of potential access to the material by malefactors, fusion has an advantage because there is no need to transport tritium beyond that needed for initial inventories of new reactors.

4. HISTORY AND PROSPECTS OF FISSION POWER SAFEGUARDS

Safeguarding of nuclear energy has been a concern since the beginning of the use of nuclear power [VIII-12]. B.M. Baruch, then ambassador of the US to the United Nations, as early as 1945-1946 proposed to make all nuclear activities, military as well as civilian, subject to international safeguards under the auspices of the United Nations. This plan failed; the US established the Atomic Energy Act of 1946 with the intention of assisting arms control by containing nuclear power as much as possible; this was a first rigid step of non-proliferation. After the success of the weapons program of the USSR, a new approach to control nuclear weapons was sought. The idea was to establish an incentive for the transfer of nuclear material away from the military domain by launching a vast program for the peaceful use of nuclear power. To facilitate this aim US Congress passed the Atomic Energy Act of 1954. Internationally, the foundation of the International Atomic Energy Agency (IAEA) in Vienna was meant to support and, at the same time, broaden this development. It was, therefore, very much consistent with this intention when the IAEA early established a safeguards system [VIII-13] which was meant to evolve with the transfer of nuclear material from the military to the civilian domain.

In the middle of the sixties, the concept of the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) arose. It is not the purpose of this discussion to elaborate on the subtle aspects of arms control and the role of non-proliferation. This has been done elsewhere [VIII-14]. Rather, the point of relevance in this context is the following: On the occasion of the NPT and its wide international negotiations, the concept and the features of the safeguards system that was meant to be an operational part of this treaty were more specifically defined. The IAEA became the institution to operate this NPT safeguards system. In 1970 and 1971 the IAEA Safeguards Committee, with the participation of about fifty nations, negotiated a document that outlines "The Structure and Content of Agreements between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons" [VIII-15]. This detailed document clarifies

the function of international NPT safeguards. The objective of NPT safeguards is defined in §28 of that document as follows:

> *... the objective of safeguards is the timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities to the manufacture of nuclear weapons or of other nuclear explosive devices or for purposes unknown, and deterrence of such diversion by the risk of early detection.*

It is important to realize the following:

(a) NPT safeguards is an agreement between governments, and the NPT as a whole is designed to avoid the proliferation of nuclear weapons on a governmental level;

(b) The NPT safeguards system concentrates on nuclear material and not on the peaceful application of nuclear power as such, because it is the nuclear material that establishes the link between the military and the civilian domains;

(c) The NPT safeguards system concentrates on the *de*tection of diversion of nuclear material and not its physical *pro*tection. This is consistent with the observation that the NPT is a treaty between states designed to avoid proliferation of nuclear weapons on a governmental level.

The essential components of the NPT safeguards system are [VIII-16]:

- materials accountability;
- containment;
- surveillance.

The emphasis is clearly on accountability. For the vast majority of nuclear facilities, namely nuclear power stations with heterogeneous cores (that is, distinct fuel elements), this is an efficient and satisfactory solution to the problem in question. This observation includes the fast breeder reactor.

Since the NPT has become effective (1969), the concern for safeguards has broadened to the extent that not only proliferation on the governmental level but also on the level of illegal groups [VIII-17] is now considered. In this case it is protection rather than detection which is of interest. In Table VIII-II we try to outline the safeguards problem accordingly.

With respect to both detection and protection, it is mostly the fuel cycle facilities other than the reactors themselves which deserve attention. Table VIII-III gives a breakdown of

Table VIII-II: Areas of Safeguards

	Detection	Protection
Governments	NPT	Addressed by the political context of the NPT
Illegal Groups	NPT	Beyond the NPT

Table VIII-III: Four Classes of Required Physical Protection

Class	Characteristics	Timing	Required Protection
1 Irradiated Material	Self-defending	To come	Very small
2 Uranium (Enrichment < 5%)	Not self-defending Enrichment required	In use	Small
3 Plutonium, U233	Not self-defending No enrichment required	To come after reprocessing	Significant
4 Uranium (Enrichment > 20%)	Ready material	No large amounts so far HTGR?	High

nuclear material in the fuel cycle. There are four classes of required physical protection. Irradiated material requires almost no protection; it is highly self-protecting, and large amounts of such material are bound to be available as more and more nuclear power stations are in operation. The second class is made up of fresh uranium with enrichment of less than 5%.

It is not self-protecting but requires enrichment for explosive applications. Fair amounts of such material are already in use. The third class consists of Pu and U233. These materials are not really self-protecting and no enrichment is needed; they therefore do require significant physical protection. The large amounts of such material come timewise after the reprocessing of irradiated material. The fourth class obviously comprises uranium enriched to 90%, which requires high degrees of physical protection.

The large fuel cycle to be envisaged if a significant share of future energy demand is met by nuclear power raises the question of the sequence of the decisions that are still pending. It is, therefore, appropriate here to consider the decision tree for the truly large scale deployment of nuclear energy. The decision tree is outlined in Figure VIII-1. The first decision

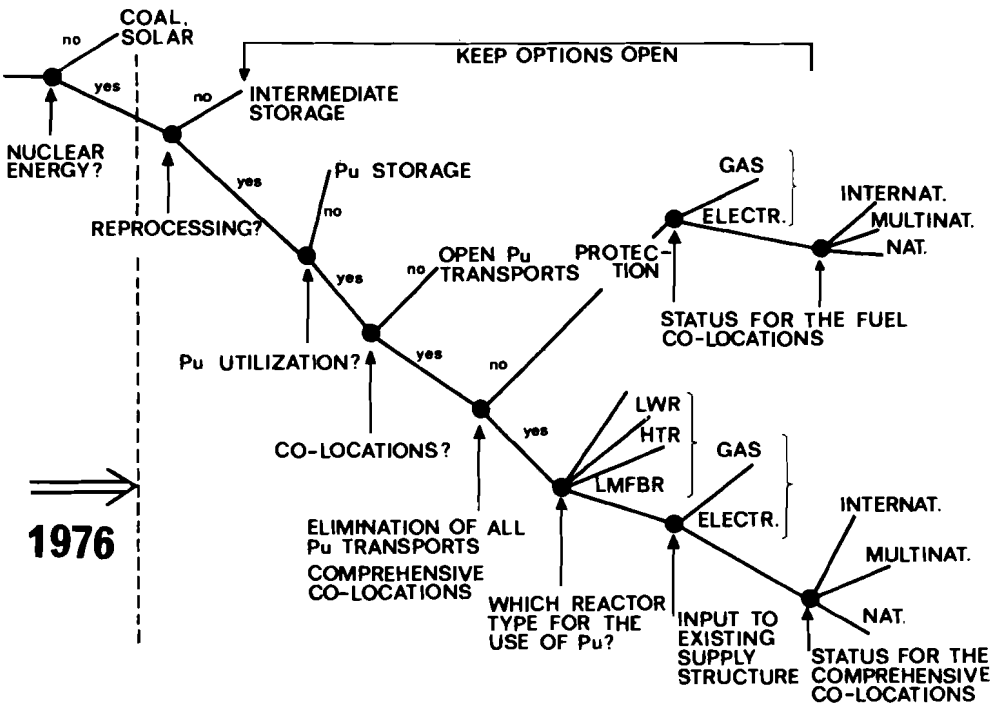

Figure VIII-1: Decision Tree for the Deployment of Peaceful Nuclear Energy

is whether or not to employ nuclear power. The next decision is whether or not to reprocess irradiated material. If not, storage that can after all be of only intermediate nature must be chosen; if yes, the decision whether or not to use Pu must be faced. If not, Pu must be stored. If for economic as well as ecological reasons this is considered to be infeasible and unwise, Pu must be used and Pu fuel elements must be fabricated. Therefore, the next question is the co-location of such Pu fuel fabrication plants with reprocessing facilities. If the answer is no, refined Pu must be transported. It is material of class 3 of the four classes of physical protection. If the answer is positive, the next question is whether or not also to eliminate the transport of fabricated (and to that extent protected) Pu fuel elements. The decision tree shown follows both options. If the decision is to transport newly fabricated Pu fuel elements (class 3), adequate physical protection must be provided for them on their way to the nuclear power stations which produce two forms of secondary energy, electricity and gas. In the long run, it may be interesting to consider the status of the implied allocation of the fuel cycle. It could be national, multinational or even international. Going back to the other branch of the decision tree, the other option would be to eliminate transport of all class 3 materials. This would necessitate the co-location not only of the fuel cycle facilities but also of reactors that are designed for the use of Pu. Then a type of reactor must be chosen that accepts the Pu produced annually.

The point to be made in this context is that the deployment of a large commercial fuel cycle for fission reactors is still to come. The application of nuclear power other than for electricity generation [VIII-18] is a possibility; accordingly, consideration must be given to the large-scale deployment of the nuclear fuel cycle [VIII-19] for fission reactors in general and fast breeder reactors in particular.

Safeguards efforts are usually underestimated, particularly those for nuclear power stations and research reactors, on both a national and an international basis. One can envisage a variety of further possibilities of safeguarding the fuel cycle and of confronting the question of physical protection more comprehensively. For example, the concept of nuclear fuel cycle parks, if accepted and properly implemented, would greatly alleviate the safeguards problem. The incentive for doing so should be large, not only because safeguards could be improved but also for reasons of the potential economical and managerial advantages and the facilitation of non-electric application. Whether safeguards will finally be considered adequate and politically satisfactory remains a problem of judgment.

5. CONCLUSIONS

Understanding and comparing safeguards aspects of fission and fusion can be facilitated by making several important distinctions:

- spread of knowledge vs. spread of nuclear material;
- explosive vs. radiological threats;
- vulnerability of power reactors vs. that of the rest of the fuel cycle;
- diversion (countered by detection) vs. theft (countered by protection);
- offenses by governments vs. offenses by private groups;
- national vs. international controls.

As fission power spreads, the associated spread of bomb-related material is more important than the spread of bomb-related knowledge. (This is a problem of all forms of fission, not just breeders.) If fusion power spreads, the associated spread of bomb-related knowledge (in the inertial confinement approach) would be more important than the spread of bomb-related material. However, neutrons from any D-T fusion reactor could be used by the operators to produce fissile material; this raises the question of "safeguarding neutrons".

Misuse of nuclear material as a radiological rather than an explosive weapon is a threat associated with both fission and fusion. (Here fusion's hazard is from the tritium associated with all approaches, not just the inertial confinement approach.) With respect to airborne dispersal of plutonium or tritium, fusion appears to have a quantitative advantage over fission of two to five orders of magnitude, depending on the chemical form of the materials; with respect to waterborne dispersal, the two are approximately even.

In the case of fission power, safeguarding other steps in the fuel cycle is a greater problem than safeguarding power stations. Moreover, the fission fuel cycle is spread out (requiring transportation of nuclear materials), and it will remain so unless nuclear energy centers become the norm. In the case of fusion, the tritium remains for the most part in the power stations; transportation is necessary only when new power stations are being started up.

The detection of diversion of nuclear material is of concern between governments, and the Non-Proliferation Treaty is addressed to this issue; it calls more for international than national controls. The protection of nuclear material from theft is a problem for individual nations and calls more for national than international controls.

For a state, the most direct route to the fabrication of a few crude nuclear explosive devices is probably the construction of centrifuges; it is not the deployment of economically significant civilian nuclear fission power. Without adequate safeguards, the latter may, however, be a permanent temptation to divert some nuclear material, possibly at a future date.

Technical, managerial and institutional measures can increase the efficiency of safeguards. Establishing an equilibrium of any kind between risks, benefits and costs is, in the first analysis, a step that entails social and political considerations as well as technical insights. The question has been raised whether adequate safeguards would entail too heavy social costs, for instance infringements of civil liberties [VIII-20]. We have not felt capable of properly addressing this issue here.

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IX. MATERIALS, AND IMPACT OF RADIATION DAMAGE

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1. FISSION REACTORS

1.1 Introduction

When the materials development for fast-breeder reactors started in the early 1960s, fuel-cladding and core-structure material were considered not to be much of a problem. LWR fuel claddings were known to undergo low-temperature embrittlement phenomena under irradiation. However, the special material phenomena occurring in a fast-reactor-core environment (high temperature, fast neutron spectrum, and high neutron fluences) were found only after a few years of operating experience with fast test reactors like RAPSODIE, DFR, EBR-II, and BR-5 in the late 1960s, and investigated thereafter.

The first small fast test reactors (Table IX-I) had been designed and built to investigate physics phenomena of fast reactors. They still had rather different fuel, clad and core-structure materials. It was not possible to do material research with them because they were not built as power reactors and thus could not reach high neutron fluences (i.e. high neutron fluxes over long time periods). Material research at first concentrated on fast-reactor fuel in thermal test reactors of the MTR and DIDO types, and in mixed-spectrum test reactors like the BR-2 (Belgium).

Table IX-I: Early Small Fast Test Reactors, Built Between 1946 and 1956

Reactor	First Operation	Power		Fuel/Clad	Coolant
		MW(th)	MW(e)		
Clementine (US)	1949	0.025	-	Pu metal, steel	Hg
EBR-I (US)	1951	1.2	0.2	U235 metal, Zry-2 clad	NaK
BR-2 (USSR)	1956	0.1-0.2	-	Pu metal, steel	Hg

About 1960 it was recognized that low fuel-cycle costs for fast reactors could be reached only with mixed-oxide PuO_2-UO_2 at burn-up rates of more than 100,000 MW(th)·day/t. The advanced fuel mixed-carbide PuC-UC was considered to become the long-run alternative because of its potential for low core inventories, high breeding ratios and, consequently, low doubling times. From 1960 onwards fast reactors of the intermediate phase (Table IX-II) were built with PuO_2-UO_2 mixed-oxide fuel, since the fabrication processes for mixed-oxide fuel were already available from LWR fuel technology. Originally, these intermediate-phase fast reactors were built and used primarily as a test bed for the statistical proof of fuel rods.

Table IX-II: Fast Test Reactors of the Intermediate Phase, Built Between 1956 and 1965

Reactor	First Operation	Power MW(th)	Power MW(e)	Fuel/Clad	Coolant
BR-5 (USSR)	1959	5	-	PuO_2-UO_2 and PuC-UC, steel	Na
DFR (UK)	1963	72	15	U235-Mo alloy, Va and Nb	NaK
EBR-II (US)	1965	62.5	20	U metal and UO_2-PuO_2, steel	Na
EFFBR (US)	1966	200	66	U235-10-Mo alloy, Zr clad	Na
RAPSODIE (France)	1967	20	-	UO_2-PuO_2, steel	Na
BOR 60 (USSR)	1969	60	12	PuO_2-UO_2, steel	Na

A new phase of material research was initiated when in 1967 irradiation from the DFR [IX-1] led to steel swelling. In the following intense research phase, it was necessary to investigate cladding and structural-steel properties such as high temperature embrittlement [IX-2, IX-3], in-pile creep phenomena [IX-4], and steel swelling as a function of temperature and neutron fluence [IX-5 to IX-9]. For porous mixed-oxide fuel then new phenomena became particularly interesting: Pu-redistribution [IX-10], density changes due to fuel sintering, irradiation-induced creep [IX-11], as well as corrosion inside the cladding by oxidation and fission-product attack, and fuel-cladding mechanical interaction at high burn-up [IX-12].

During this material development phase, it was also found that prototypical-material test parameters for the cladding and other core-structural materials cannot be fully reached in

small fast test reactors. Such test conditions will only be achieved in prototype fast-power reactors of the 300 MW(e) class, and later in large commercial LMFBRs (Table IX-III). One understands why by comparing the maximum fuel burn-up and the neutron fluences to the cladding for three different reactor sizes, as given in Table IX-IV. Whereas a prototypical fuel burn-up can be reached in small fast test reactors (RAPSODIE Fortissimo), the high neutron fluence of $3 \cdot 10^{23}$ n/cm^2 is only reached in SUPERPHENIX, with PHENIX representing an intermediate step.

1.2 Status of Fuel Development

Design studies and construction experience with prototype fast reactors result in reactor material-core compositions of about 30 per cent fuel (fissile and fertile), 20 per cent steel (clad, spacers, and core structure), and 50 per cent coolant (see Chapter IV). The fuel contains about 12 to 35 per cent plutonium (fissile) mixed with U238 (fertile material). Plutonium enrichment depends upon the size of the core (i.e. the criticality conditions). Usually different radial enrichment zones in the core are used in order to obtain a flat radial power profile over the core. If the fuel is reprocessed in a pure fast-reactor fuel cycle, the Pu-isotopic analysis is 75 per cent of Pu239, 22 per cent of Pu240, 2.5 per cent of Pu241, and 0.5 per cent of Pu242. It differs slightly if the fuel is taken from an LWR fuel cycle.

With these basic characteristics the specific power of the fast reactor core reaches a level of 0.7 to 1 MW(th)/kg fissile material. The fuel is filled into stainless-steel tubes in the form of cylindrical pellets. Before the end caps of both ends of the canning tubes are welded, helium is filled in to increase the heat transfer between fuel pellets and steel cladding. Fuel pellets are fabricated in a pressing and sintering process which--because of α-radiation and radiotoxicity of plutonium--must take place in leak-tight boxes. Each fuel rod contains a pellet section with fissile material, and two sections with fertile fuel (U238). At the upper or lower end of the fuel rod, gaseous fission products are collected in a fission gas plenum. Gas pressures up to 70 atm can be reached within the fuel rod at the end of the irradiation time. Each fuel element contains between 61 and 323 fuel rods, held either by spiral wires or spacer grids within the fuel element box.

In order to select a suitable fuel alternative for fast-reactor cores, the following criteria have to be discussed:

- fuel density;
- melting point, thermal conductivity, heat capacity, and attainable specific fuel-rod power;
- fuel crystallographic structure, thermal expansion, and the Doppler coefficient (for safety reasons);

Table IX-III: Prototype Fast Power Reactors

Reactor	Country	Current Status	Power (MW(e))	Fuel Enrichment (%)	Material Fuel/Clad	Coolant	Fuel Burn-up MW(th)·day/t
PHENIX	France	In operation since 1973	250	17/24	PuO_2-UO_2, SS 316	Na	50,000
PFR	UK	In operation since 1975	254	19/25	PuO_2-UO_2, SS 316	Na	70,000
SNR 300	FRG[a]	Start of operation 1982	282	25/35	PuO_2-UO_2, SS 1.4970	Na	55,000
CRBR	US	Start of construction 1977	360		PuO_2-UO_2, SS 316	Na	70,000
MONJU	Japan	Start of construction 1978	300		PuO_2-UO_2, SS 316	Na	80,000
BN 350	USSR	In operation since 1973	350[b]		PuO_2-UO_2, SS	Na	50,000
BN 600	USSR	Start of operation 1977	600		PuO_2-UO_2, SS	Na	90,000
SUPERPHENIX	France	Start of construction 1976	1200		PuO_2-UO_2, SS 316	Na	70,000
CFR-1	UK	Start of construction 1978	1250		PuO_2-UO_2, SS	Na	>70,000

[a] together with Belgium and the Netherlands;

[b] or 150 MW(e) and 120,000 m^3/day freshwater.

Table IX-IV: Comparison of Maximum Fuel Burn-up and Total Neutron Fluence to Clad, for Three Different Sizes of Fast Power Reactors: RAPSODIE Fortissimo, PHENIX, and SUPERPHENIX

Fuel Element Design Data		SUPERPHENIX (as of Sept. 1974)	PHENIX	RAPSODIE Fortissimo
Number of pins per fuel element		271	271	61
Clad material		SS 316	SS 316	SS 316
Clad outer diameter	mm	8.50	6.55	5.10
Gas plenum (length)	mm	lower (850)	lower (440)	lower (100)
Pin length	mm	2700	1793	531.5
Max. linear power	W/cm	450	430	400
Max. clad temperature	°C	620	650	650
Nominal burn-up	MW(th)·day/t	70,000	50,000	65,000
	%	8	6.4	8.1
Total fluence at nominal burn-up	10^{23} nvt	3	2	0.65

- fission gas and swelling behavior;
- maximum attainable burn-up;
- compatibility with steel cladding;
- fuel fabrication process (in economic terms), and reprocessing of irradiated fuel.

Tables IX-V, IX-VI, and IX-VII compare some of the typical properties of metallic and ceramic U-Pu fuel alternatives [IX-24]. Uranium-plutonium metal has a high density and a good thermal conductivity, but its melting point is relatively low. In addition, several of its metallic phases are in the temperature range which can occur in fast-reactor cores (the coolant temperatures are between 350 and 550°C). For lack of microstructural stability and because of heavy fission gas swelling, U-Pu metal cannot reach the required high burn-ups.

Table IX-V: Characteristic Properties of Metallic and Ceramic Fuels [IX-13 to IX-18]

	U Metal	Pu Metal	UO_2	PuO_2	UC	PuC	UN	PuN
Theoretical Density (g/cm^3)	19.04-18.0, 3 metallic phases	19.86-16.51, 6 metallic phases	10.96	11.46	13.63	13.60	14.32	14.25
Melting Point (°C)	1132	640	2800	2290	2400	1650	2850	2750
Expansion Coefficient ($10^{-6}/°C$)	α = 17 β = 22 γ = 17	α = 67 β = 41 γ = 35 γ = -8.5 γ = -596 ε = 15	10.5	11.4	11	10.8	10.1	11.2

Table IX-VI: Thermal Conductivity and Maximum Fuel Rod Power for Metallic and Ceramic Fuels [IX-19, and IX-20]

	U-Pu Metal	UO_2-PuO_2	UC-PuC	UN-PuN
Relative Thermal Conductivity (UO_2-PuO_2 conductivity = 1)	12	1	7	7
Maximum Fuel Rod Power (W/cm fuel length)	1300	550	2300	1900

Table IX-VII: Compatibility of Ceramic Fuel with Na and Steel at Zero Burn-up [IX-21 to IX-23]

	UO_2-PuO_2	PuC	UC	UN, PuN
Compatibility with Na	Good up to 680°C	Good up to 430°C	Good up to 600°C	Good up to 800°C
Compatibility with Steel	Good up to 1400°C	Reaction at 1000°C	Reaction at 1000°C	Good up to 1000°C

Only ceramic fuel can be irradiated to a burn-up of more than 100,000 MW(th)·day/t. As compared to PuO_2-UO_2, both PuC-UC and PuN-UN have higher densities, better fuel conductivities,

Table IX-VIII: Fuel Pin Data for Sodium Cooled Fast Power Reactors [IX-25]

		USA			USSR	
		EBR-II Test S/A	FFTF	CRBR	BOR 60	BN 350
Reactor Power	MW(th)	62.5	400	950	60	1000
Fuel Pin Diameter	mm	5.84	5.84	5.84	6.1	6.1
Total length	mm		2380	2911	1100	1140
Fuel Material		UO_2-PuO_2 [a]	UO_2-PuO_2	UO_2-PuO_2	UO_2 [a]	UO_2 [a]
Length	mm	343	914	914	400	1060
Diameter	mm	4.94	4.94		5.1	4.82
Bulk density	%th.d.	90.4	90.4		93.5	95.8
Smeared density	%th.d.	85.5	85.5			73.0
Cladding Material		SS 316	SS 316	SS 316	SS 316	SS 316
Wall Thickness	mm	0.38	0.38	0.38	0.40	0.35
Linear Rod Power, max. [b]	W/cm	450/	425/	475/	550/	440/500
Cladding Temperature, max. [b,c]	°C	630/	610/750	640/	/800	
Burn-up, Local, max.	MW(th)·day/kg	80	80	150	100	50
Pins Per Subassembly Fuel Element		61	217	217	37	169
Pitch to Diameter Ratio				1.25	1.10	1.15
Spacer Type			Wire	Wire (Grid)	Wire	Wire

and the potential for higher fuel-rod powers. However, the fuel fabrication process for PuC-UC is not yet fully developed. More intense PuC-UC irradiation programs will be started in the coming years. So far, still much less research and development work has been done for PuN-UN.

On the other hand, already at the beginning of the fast-reactor fuel development, PuO_2-UO_2 benefited from the experience in a well-known fabrication and reprocessing process which had been gained with the LWR fuel technology. Its physico-chemical properties are good enough to guarantee economical fuel-cycle costs and a safe core behavior. From 1960 onwards, therefore, all fast reactor project groups used PuO_2-UO_2 mixed oxide as reference fuel (Tables IX-III and IX-VIII).

USSR	France			UK		
BN 600	RAPSODIE Fortissimo	PHENIX	SUPER-PHENIX	DFR Test S/A	PFR	CFR-1
1480	40	563	2910	72	600	2900
6.9	5.1	6.55	8.65	5.84	5.84	5.84
2445	532	1793		532	2260	2112
UO_2[a]	UO_2-PuO_2[a]	UO_2-PuO_2	UO_2-PuO_2	UO_2-PuO_2[a]	UO_2-PuO_2	UO_2-PuO_2
750	322	850	≈1000	304	914	1000
5.9	4.23	5.50	7.0			
95.2		85				
				80	80	
SS 316	SS 316	SS 316L		SS M316	SS M316	
		0.45		0.38	0.38	
530/690	430/	430/	450/	≈430/	450/	
/690	650/710	640/690	620/700	/700	/700	/700
100	60	60	100	100	75	100
127	61	271	271	77	325	325
1.17				1.26	1.26	1.26
Wire	Wire	Wire		Grid	Grid	Grid

Table IX-VIII: Fuel Pin Data for Sodium Cooled Fast Power Reactors [IX-25] (Cont'd)

		FRG with Belgium and the Netherlands				Italy
		KNK-II Test S/A	SNR 300 MARK-Ia	SNR 300 Mark-II	SNR-2	PEC
Reactor Power	MW(th)	58	736			130
Fuel Pin Diameter	mm	6.0	6.0	7.6	7.6	6.7
Total length	mm	1541	2475	2475		
Fuel Material		UO_2-PuO_2 [a]	UO_2-PuO_2			UO_2-PuO_2 [a]
Length	mm	600	950	950	1200	650
Diameter	mm	5.09	5.09			
Bulk density	%th.d.	86.5	86.5	86.5		
Smeared density	%th.d.	80	80.0	80.0		
Cladding Material		SS 1.49..	SS 1.4970	SS 1.4970		SS 316
Wall Thickness	mm	0.35	0.38	0.50	0.50	0.40
Linear Rod Power, max.[b]	W/cm	460/570	370/455	440/540	450/	390/
Cladding Temperature, max.[b,c]	°C	/670	620/670	620/670		640/
Burn-up, Local, max.	MW(th)·day/kg	90	90	90	115	
Pins Per Subassembly Fuel Element		211	169	127	271	91
Pitch to Diameter Ratio		1.32	1.32	1.16	1.16	1.18
Spacer Type		Grid	Grid	Grid (wire)		

1.3 Behavior of Mixed Oxide Fuel Under Irradiation

At high burn-up and high specific power the PuO_2-UO_2 mixed-oxide fuel considerably changes its physico-mechanical and chemical behavior. As shown in Figure IX-1, a central channel and several radial cracks build up after irradiation at high specific power. At the same time radial density changes occur within the pellet. Two competing effects act on the pellet diameter: the pellet diameter decreases with the sintering effects within the pellet by induced irradiation [IX-26 to IX-28]; and fuel swelling continuously increases the diameter, so that the limited sintering effect is overcome. Fuel swelling is caused by solid fission products (mainly cesium), and also by gaseous fission products (xenon and krypton), which in the form of gas bubbles partly remain in minute pores or in the crystal lattice of the fuel [IX-29 to IX-31]. Depending upon fuel temperature and burn-up, almost all of the gaseous fission products are released either into larger open fuel spaces (cracks

Japan		India
JOYO	MONJU	FBTR
100	714	
6.3	6.5	5.1
1910	2740	532
UO_2-PuO_2[a]	UO_2-PuO_2	
600	900	
5.4	5.4	
93.5	85	
SS 316	SS 316	SS 316
0.35	0.45	
	457/	
	80	
91	169	61
1.21		
Wire		Wire

[a] Uranium enriched in U235;
[b] Nominal condition/hot-spot condition;
[c] In some cases cladding midwall temperature.

or central channel), or into the fission gas plenum acting as internal pressure on the cladding. High contact pressures between fuel pellet and cladding rarely develop under steady-state operating conditions because of irradiation-induced creep effects within the fuel pellet. However, it seems that strong mechanical interaction between fuel and clad can occur during fuel-rod power changes, and perhaps also if volatile fission products locally accumulate at high burn-ups [IX-32].

The chemical behavior of irradiated mixed-oxide PuO_2-UO_2 fuel is determined by the fact that fissioned PuO_2 releases oxygen and generates fission products. These phenomena occur in a very steep radial temperature gradient of the pellet, which finally leads to both a higher oxygen potential within the pellet and space-dependent concentrations of plutonium, oxygen and fission products (thermo-diffusion). Among the solid fission products, it is mainly cesium which can attack the cladding by inside corrosion [IX-32 to IX-36]. This inside-corrosion process depends on the temperature and is of importance

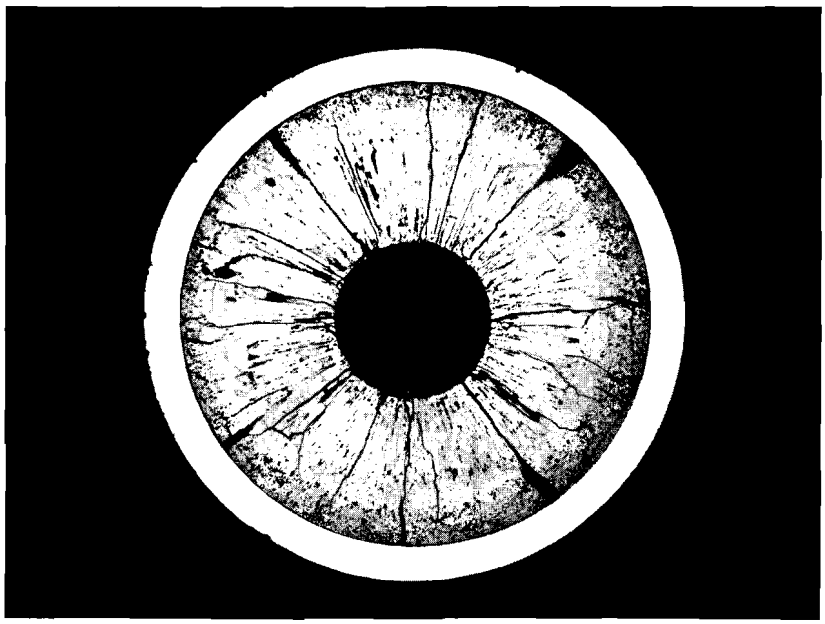

Figure IX-1: Fuel Structure in a PuO_2-UO_2 Fuel Rod After a Burn-up of 5% at a Rod Power of 500 W/cm

at temperatures above 600°C. Such temperatures are usually reached at the upper end of the fuel pins with the highest rate.

All these physico-mechanical and chemical effects have been measured in special fuel-irradiation experiments in every fast-reactor project. The experimental data available form the basis of the fuel-pin design and of specifications for the fabrication process. Many mixed-oxide fuel pins have already been irradiated up to between 120,000 and 170,000 MW(th)·day/t in irradiation experiments in RAPSODIE, EBR-II, DFR and BOR 60 [IX-37 to IX-41]. In the first cores of PHENIX, the French prototype fast reactor, and the Russian BN 350, the highest rated fuel pins have up to now reached burn-ups of 60,000 MW(th)·day/t without any fuel-pin failure. It is expected to have higher burn-ups after there has been more operating experience with these prototype power reactors.

1.4 Behavior of Mixed Oxide Fuel Under Off-normal Operating Conditions

For considerations related to safety design, one must know the behavior of fuel rods under off-normal operating conditions. Off-normal operating conditions of a fast reactor core are

described as cyclic power changes, power transients, and loss of cooling conditions. Within the last five years, each of these problem areas has been covered and investigated by special experimental programs in research reactors like TREAT in the US [IX-42], CABRI in France [IX-43], and HFR in the Netherlands [IX-44]. It has been possible to measure the behavior of the fuel even under burst or melt-down conditions, as well as thermal interaction phenomena of molten fuel with Na as coolant. These data are the basis of licensing procedures for prototype fast reactors. Special fuel-safety experiments will be extended within the next ten years, so that comprehensive safety information for the licensing of large commercial fast reactors will be available around 1985 to 1990.

1.5 Development of Mixed Carbide PuC-UC Fuel

As can be seen from Tables IX-V and IX-VI, Pu-U monocarbide has a higher fuel density and a better thermal conductivity than PuO_2-UO_2. These properties allow for a higher rod power of the fuel pin, higher breeding ratios, a lower fuel inventory, and lower doubling times for fast reactor cores [IX-45].

Mixed PuC-UC fuel is, therefore, considered an alternative advanced fuel for future fast breeders. Fuel fabrication and irradiation programs are pursued by all fast-reactor projects.

With a higher fissile density and a lower operational temperature, irradiation-induced swelling of PuC-UC fuel has been found to be relatively high. Inside-corrosion phenomena by fission products as found in PuO_2-UO_2 fuel rods are not expected [IX-46].

Fabrication processes for PuC-UC fuel and fuel rods are still being developed. He-bonded and Na-bonded fuel and fuel rod concepts are actively tested in experimental programs [IX-47 to IX-49]). For a number of carbide fuel rods, burn-up rates up to 10 per cent have already been obtained without failure. However, reprocessing for the PuC-UC fuel still has to be developed. Most of the R&D programs for the PuC-UC fuel, therefore, follow schedules by which carbide fuel element concepts would be provided by 1985 to 1990.

1.6 Status of Cladding Development

The fuel cladding of 0.4 to 0.5 mm thickness and an outer diameter of 6 to 8.5 mm (Table IX-VIII) can be considered the most strained part of the fast-reactor core. In a temperature range of 350°C to about 700°C, it is subjected to internal mechanical loads by fission-gas pressures and fuel swelling, inside-corrosion attack by fission products, and external corrosion due to sodium impurities. Most of these phenomena are a function of temperature and irradiation time (neutron fluence $>10^{23}$ n/cm^2) (see Figure IX-2) [IX-12, IX-50, and IX-51].

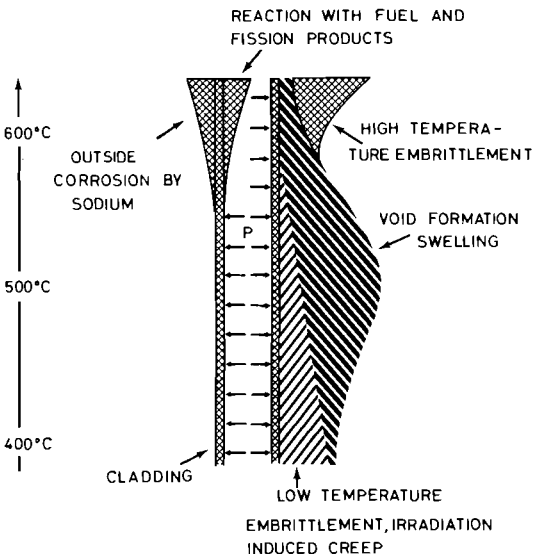

Figure IX-2: Irradiation Damage and Corrosion Effects Acting on the Fast Reactor Fuel Cladding

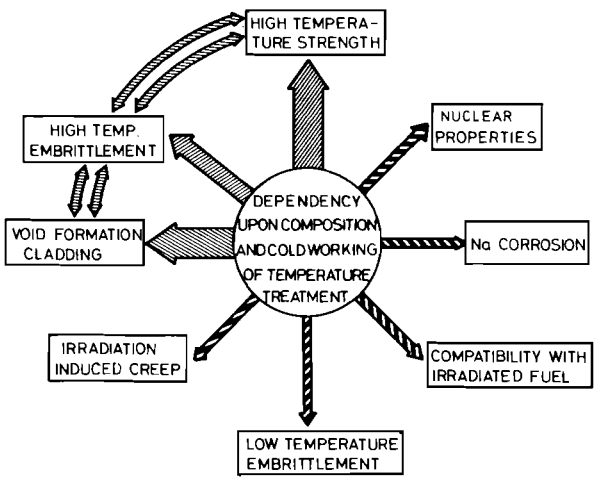

Figure IX-3: Schematic Diagram of the Dependency of Various Phenomena and Steel Properties Upon Steel Composition, Cold Working, and Temperature Treatment

As can be seen from Figure IX-3, the favorite cladding material must have an optimal combination of required properties. This narrows the choice among the different materials available to a very small selection window.

When the development of fast-reactor fuel cladding began in the early 1960s, the main selection criteria were:

- a small neutron absorption cross-section to keep the breeding ratio high;
- sufficient corrosion resistance against the coolant, sodium;
- good creep-rupture properties at high temperatures;
- good weldability.

Table IX-IX shows the absorption cross-sections for metals and alloy elements of interest at 0.1 MeV in a fast neutron spectrum [IX-52]. The various metals and elements are subdivided

Table IX-IX: Absorption Cross-sections for Metals and Alloy Elements

Group	Metal or Alloy Element	$\sigma_{n\gamma}$ (100 keV) (mb)
1	Al	4
	Ti	6
	Fe	6.1
	Cr	6.8
	V	9.5
2	Si	10.0
	Co	11.5
	Ni	12.6
	Zr	15.1
	Cu	24.9
	Mn	25.6
3	Mo	71.0
	Nb	100.0
	W	178.0
	Ta	325.0

into three groups. For fast-reactor cores an absorption cross-section of about 10 mb is considered acceptable, which narrows the selection down to *Group 1* and half of *Group 2*. Out of the remaining metals only iron (austenitic steels), nickel (nickel-base-alloys), and vanadium (vanadium-base alloys) are of relevance here. Aluminum does not have sufficient high-temperature strength properties. Chromium or chromium alloys are too brittle, and cobalt must be excluded because of its Co60 isotope (which has a high γ-activity, and a long lifetime).

Table IX-X and Figure IX-4 compare the absorption cross-sections and creep-rupture-stress properties of austenitic steels, nickel-base alloys, and vanadium alloys [IX-51, IX-52]. In all three groups alloys can be found whose required creep-rupture-stress properties were above 10 to 15 kg/mm^2 at 650^{0}C. Some of the nickel-base alloys and also of the vanadium alloys have a better high-temperature strength than austenitic steels. But they also have higher absorption cross-sections, which would decrease the breeding ratio. The development of vanadium alloys is still in its very initial phase. The nickel-base alloys do not have the same high corrosion resistance against sodium with its impurities (O_2, H_2) as austenitic steels. Considerable mass-transport phenomena from hot to cold parts of the primary-cooling circuits of LMFBRs have been found with nickel-base alloys [IX-53]. In addition, some of the nickel-base alloys have shown a higher tendency to high-temperature embrittlement, which is caused by He-bubble formation at grain boundaries [IX-12, IX-51].

For these and a number of other reasons, all fast-reactor projects in the US, the UK, France, the FRG together with Belgium and the Netherlands, Japan, and the USSR have chosen austenitic steels as reference material for the clad and other core-structural materials. They have been found to represent the best alternative for the required combination of properties shown in Figure IX-3.

Whereas in the US, the UK, France, and Japan unstabilized steels of the type SS 316 are preferred, the German/Belgian/Dutch SNR project chose a titanium-stabilized austenitic steel as reference material for the cladding, and a niobium-stabilized austenitc steel for other core-structural material. Stabilized austenitic steels are also used in the USSR.

Since 1967, when British experts reported about the irradiation behavior of steel under high neutron doses [IX-1], the problem of steel swelling has been actively studied. Steel swelling is caused by irradiation damage (void formation by vacancy condensation) at neutron fluences beyond 10^{22} nvt. It depends upon the irradiation temperature and the neutron fluence (see Figure IX-5 and IX-6), and can lead to a high increase in volume [IX-12, IX-50, IX-54]. This has to be accounted for when clearances and tolerances are fixed in the design of fuel elements and the entire core structure. Steel swelling is also dependent upon the composition (i.e. the alloy components) of

Table IX-X: Material Composition of Various Steel and Absorption Cross-sections for Fast Neutrons

	Alloy	Compositions (%) C	Fe	Cr	Al	Ti	V	Si	Ni	Mn	Mo	Nb	W	Absorption Cross-section for 0.1 MeV Neutrons (mb)
Austenitic Steels	X5 CrNi 18 9/AISI 304	0.05	*	18.0				1.0	9.0	2.0				7.25
	1.4981	0.06	*	16.0					16.0		1.8	0.7		8.50
	1.4970	0.10	*	15.0		0.4			15.0		1.2			8.40
	X8 CrNi MoV Nb1613/1.4988	0.08	*	16.0			0.75	0.5	13.0	1.0	1.2	1.2		8.34
	X5 CrNiMo 18 12/AISI 316	0.05	*	18.0				1.0	13.5	2.0	3.0			8.50
	Incoloy 800	0.1	*	23.0	0.6	0.6		1.0	35.0	1.5				8.73
	Incoloy TYP	0.1	*	23.0	0.6	0.6		1.0	35.1	1.5	5.0	5.0	5.0	15.49
Ni Alloys	Inconel 600	0.15	10.0	17.0				0.5	*	1.0				10.94
	Inconel X 750	0.08	9.0	17.0	1.0	2.75		0.5	*	1.0		1.2		11.41
	Hastelloy X	0.1	18.5	22.0				0.75	*	1.0	9.0		0.6	13.22
	Inconel 718	0.1	*	21.0	0.8	1.15		0.35	55.0	0.35	3.3	5.5		14.14
	Inconel 625	0.1	5.0	23.0	0.4	0.4		0.5	*	0.5	10.0	4.15		17.06
V Alloys	V-20 Ti					20	*							8.71
	V-20 Ti-10 Nb					20	*					10		14.48
	V-10 Ti-15 Nb					10	*					15		17.10

* balance

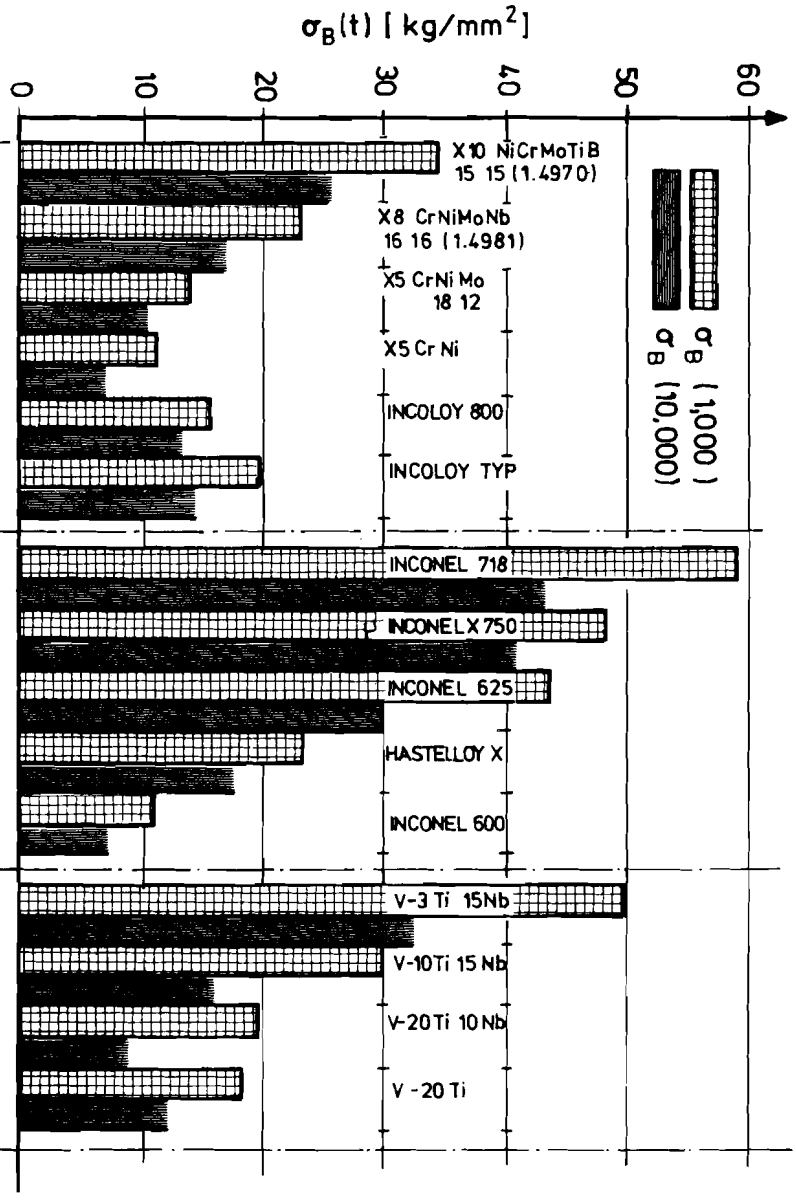

Figure IX-4: Creep Rupture Stress $\sigma_B(1000)$ and $\sigma_B(10,000)$ at 650°C

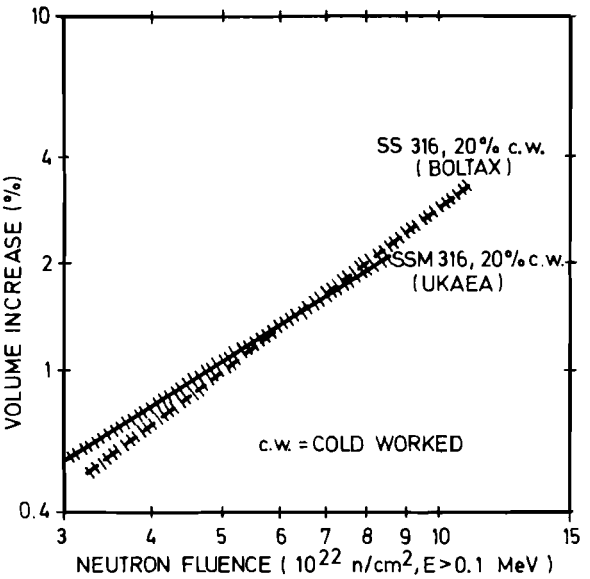

Figure IX-5: Volume Increase of Steel (Swelling Effect) as a Function of Neutron Fluence (Irradiation Temperature Constant at 550°C)

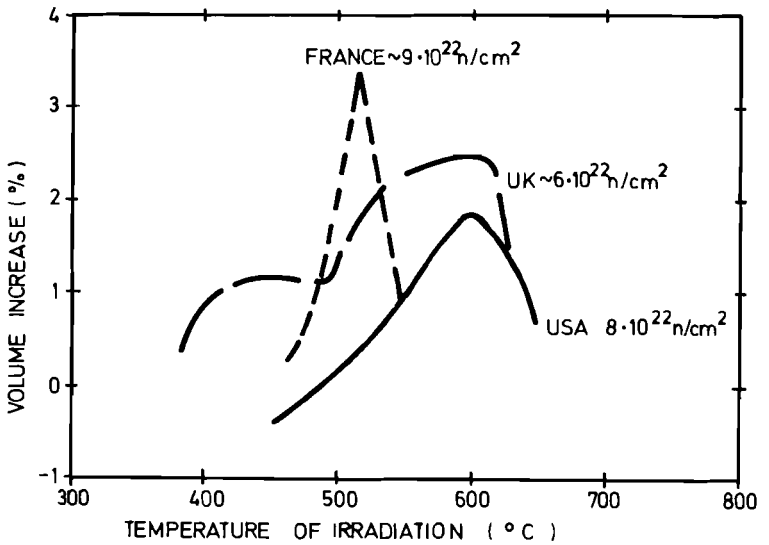

Figure IX-6: Volume Increase of Type 316 Stainless Steel (Swelling Effect) as a Function of Irradiation Temperature

the austenitic steel in question (see Figure IX-3), and can be decreased by 20 per cent cold working during fabrication of the cladding tubes. This also increases the high-temperature strength properties of some austenitic steels such as, for instance, the titanium-stabilized austenitic steel used in the SNR project.

Unfortunately, there is still another irradiation effect-high-temperature embrittlement--that must be accounted for [IX-51, IX-55]. High-temperature embrittlement is mainly caused by the formation of He-bubbles at grain boundaries when (N,α) reactions with different alloy elements occur at neutron fluences beyond 10^{22} nvt. Another initiating effect is radiation damage by displacements. It is mainly effective above temperatures of 600°C. High-temperature embrittlement becomes worse when cold working of the cladding is applied. However, this can be balanced by a temperature treatment (annealing) of the steel tubes after fabrication and cold working [IX-51].

Two other irradiation-damage effects have been found: low-temperature embrittlement, and irradiation-induced creep. Low-temperature embrittlement starts at very low radiation doses (approximately 10^{17} n/cm^2). It increases with increasing neutron dose and obviously reaches saturation at neutron doses of about 5 to $10 \cdot 10^{22}$ n/cm^2, depending upon irradiation temperature. Irradiation-induced creep phenomena can also lead to dimensional changes of the cladding or other core-structure parts. Both phenomena occur below 550 to 600°C and are not easy to improve by changing the alloy composition of austenitic steel nor by cold working. Irradiation-induced creep is linearly dependent on stress and is obviously correlated to swelling. Both phenomena have to be accounted for in calculating the cladddding thickness (internal fission-gas pressure) and the design of the fuel element boxes. The lower parts of the fuel element boxes undergo high stresses due to inside coolant pressure, thermal expansion, and thermal gradients in the fast-reactor core [IX-50].

Outside corrosion at the austenitic steel cladding has been measured to be dependent upon sodium temperature, sodium velocity, and oxygen content. Sodium corrosion acts on the outside of the cladding by dissolving some of the alloy components of the austenitic steel from a thin surface layer at corrosion rates of about 10^{-3} cm/yr [IX-56 to IX-58]. These radioactive corrosion products can be transported within the primary coolant circuits to colder parts in the heat exchangers, where they may render repair and maintenance work more difficult [IX-59].

In summarizing the present status of steel development for fast-reactor cores it can be stated that: All fast-reactor projects in the various countries have collected the necessary data for material damage phenomena induced by irradiation. Reference steels are either nonstabilized or stabilized

austenitic steels with 20 per cent cold working and subsequent temperature treatment. The operation of the prototype fast-reactors, especially of PHENIX in France and the BN 350 in the USSR, has provided experience with the selected reference steels at neutron doses up to 10^{23} n/cm^2.

Within the next few years much more data will become available for higher neutron doses and for alternative steels in reactors such as PFR, BN 600, SNR 300, etc. The future development will aim at alternative steels of swelling rates and neutron-induced creep rates that are still lower, as well as for even better high-temperature strength properties. This development is necessary since future large commercial LMFBRs of more than 2000 MW(e) require maximum neutron dose rates that are almost double those of today's prototype LMFBR (see Figure IX-7). As regards alternative cladding and structural materials, ferritic steels, Fe-Cr-Ni (a nickel-base alloy), and vanadium alloys perhaps will have to be investigated.

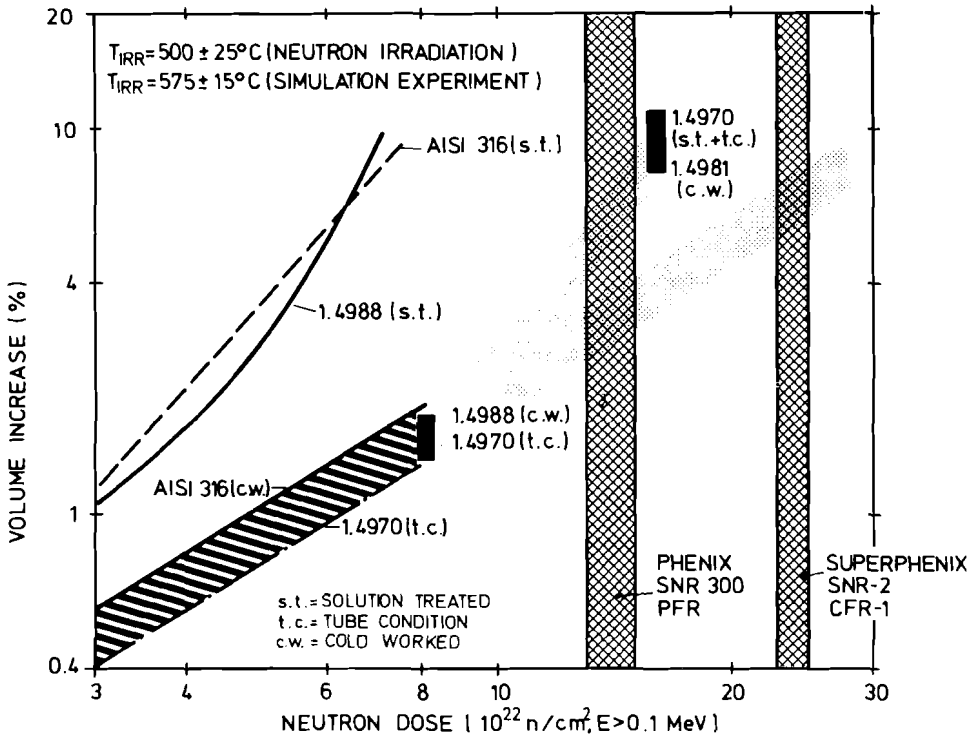

Figure IX-7: Swelling Behavior of Various Steels and Irradiation Doses Reached in Different LMFBRs

1.7 Materials for LMFBR Heat Transfer Systems

For reactor vessels, primary and secondary coolant circuits, heat exchangers, valves, etc., an austenitic steel of the AISI 304 type is used in LMFBRs. Reactor-vessel steel is exposed to maximum neutron fluences of 10^{19} n/cm^2, which leads to a reduction of the time-to-rupture properties of this steel. The other parts of the plant are hardly subject to radiation damage. Problem areas are mainly high-temperature strength and creep data up to temperatures of 550°C for normal operation, and up to 850°C for special safety considerations. Three-dimensional stress problems for valves or other components have to be handled for temperature shock and other ASME case conditions [IX-60].

Respective R&D programs for the collection of these data have already been carried out, or are still underway.

2. FUSION REACTORS

2.1 General Requirements

In contrast to fission reactors where there are perhaps four or five different classes of *solid* materials to worry about (fuel, cladding, and core restraint, reflectors, control rods, and pressure vessels), there may be as many as ten in fusion reactors. These general classes are listed below:

- low Z liners;
- electrical insulators;
- structure;
- solid tritium breeder material*;
- fissile breeder material*;
- neutron multipliers*;
- reflectors;
- shielding material;
- magnets*; and
- optical systems*.

The functions of these materials have been reviewed before [IX-61 to IX-63], and we will only briefly summarize the results here.

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* only required in some systems.

TOKAMAKs may require low-atomic-weight liners for plasma-physics reasons (mainly to insure that excessive power is not leaked from the plasma by radiation from impurity atoms) [IX-64, IX-65]. Such materials are placed in the vacuum region between the plasma and the first solid wall, and therefore will be subjected to the most extreme temperature, charged-particle, photon, and neutron environments in the reactor. It is important that they last at least as long as the first structural walls in order to allow for some chance of economical commercial operation.

Electrical insulators will be required in all magnetically confined systems in one form or another, and in inertially confined systems using electron beams. Both Mirrors and TOKAMAKs will require fueling systems (either beams or pellets) which will probably rely on electrostatic acceleration. It will be difficult to shield such systems from the direct bombardment of neutrons and the associated damage to dielectric properties. Auxiliary heating in TOKAMAKs may utilize radiofrequency sources, which could contain dielectrics to reduce the size of the waveguides [IX-66]. Unfortunatley, such waveguides must "see" the plasma, and therefore be subjected to all the radiation emanating from the plasma. Theta Pinch reactors will require electrical insulators which can maintain dielectric strengths of up to 100 kV/cm while being pulsed some 3 to 10 million times per year [IX-67]. This insulator must function at temperatures approaching $1000^{0}C$ while being bombarded with copious amounts of atomic hydrogen and neutron fluences of up to 10^{22} n/cm^{2} per year. Finally, the electrodes of E-beam reactors will be subjected to very high fluxes, and the associated degradation of properties has not been completely appreciated at this time.

The needs for structural vessels which provide vacuum tightness and are pulsed anywhere from 10 to 10^{8} times per year are fairly well established and extremely demanding. Reflectors are required for efficient neutron utilization and protection of components outside the reactor. Shields are absolutely essential to prevent excessive radiation levels from occuring outside the reactor, and to prevent damage to components not directly involved in the extraction of energy (i.e. magnets, lasers, fueling devices, etc.).

Breeding is, of course, absolutely essential for all the D-T systems. If liquid lithium is not used, then solid lithium containing compounds as Li_2O, Li_7Pb_2, $LiAlO_3$, or $LiSiO_2$ must be utilized [IX-68, and IX-69]. We will see later that these compounds have many of the same problems now facing us for fission reactor fuels with a few added features which could be quite difficult to overcome. For example, these materials have high volumetric heating rates, which promote high temperatures and severe temperature and swelling gradients in ceramics. In order to avoid large tritium inventories, the tritium must be constantly removed from the breeder. If radiation damage and high temperatures interfere with the diffusivity and/or the diffusion path of the tritium, very high tritium inventories could occur.

The use of solid breeding compounds will almost invariably require the use of neutron multipliers such as beryllium*. Ignoring the high cost per kg of Be and the lack of extensive Be reserves in the world for the moment [IX-70], one finds that severe dimensional problems could occur at high temperature due to high helium-gas generation rates. It has even been proposed to surround the plasma with a blanket containing fissionable and/or fertile material [IX-71]. Presently it seems that the latter is more attractive, and this would mean that the associated dimensional problems of U and Th compounds need to be considered, along with radioactive fission products and associated safety questions.

The critical properties of superconducting magnets for TOKAMAKs and Mirrors and of conventional magnets for Theta Pinches must be maintained. In the first case, the critical currents, temperatures, and fields of superconducting filaments must not be significantly reduced over long time periods by bombardment with neutrons. The cryogenic stabilizers must retain a low resistivity value to insure safe operation in the event of a quench in the superconductor. The compression coils of a Theta Pinch reactor will be subjected to high neutron fluxes and operate above room temperature. Significant transmutation reactions and displacement rates will result in changes in resistivity over long periods of time. Finally, one must be concerned about the non-current carrying components of magnets, such as electrical insulators or thermal insulators. Failure of any of these components could require costly repairs and unacceptable down times.

The optical systems for laser reactors (such as Mirrors and Windows) must maintain extreme dimensional stability (to within a quarter of the wavelength or about 3000 Å) over long periods of time in a commercial system. Very little is presently known about the effects of neutron damage on the surface roughness, and the subsequent effects on the efficiency of pellet compression, or the rate at which implosions can take place. This area is almost completely devoid of prior information, and it is difficult to even *estimate* how serious a problem may be encountered. In the worst case, the ultimate viability of a laser reactor could critically hinge on the successful solution of such problems.

An attempt to summarize the preceding discussion has been made in Table IX-XI, where the potential materials, their functions, and their critical properties are listed. A detailed discussion of these numbers is not warranted here because the quantitative numbers are somewhat design-dependent. However, the reader will note the wide range of materials involved in fusion systems, and he can draw his own conclusions about the extent of the research problem to establish engineering feasibility of CTR reactors.

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* See Chapter V.3.

Table IX-XI: Summary of Materials Requirements in the Nuclear Island Portion of D-T Fusion Reactors

Primary Function	Main Requirements [a]	Principal Candidates	Typical Operating Temp. (°C)	Approximate Maximum Neutron Flux ($n/cm^2 \cdot yr$ per MW/m^2) [b] 14.1 Mev	Total
Structural	High temperature strength Acceptable heat-transfer properties Low macroscopic absorption cross-section Low induced radioactivity Compatible with coolant Fabricable Readily available, low cost Some resistance to radiation damage Non-magnetic (except for laser reactors)	Al alloys SS Nickel-based alloys V alloys Nb alloys Mo and Mo alloys	<300 <650 <700 <800 <1000 <1000	$1.4 \cdot 10^{21}$	$7 \cdot 10^{21}$
Coolants	High heat capacity High thermal conductivity Low viscosity Low pumping power (especially in magnetic fields) Low macroscopic absorption cross-section	He Li Na Li salts K	Depends on structure	$1.4 \cdot 10^{21}$	$7 \cdot 10^{21}$
Breeding	High tritium production cross-section Reasonable tritium extraction Low T_2 inventories	Li Li_7Pb_2 LiAl $LiAlO_2$ LiO_2	<1000 <300 <500 <600-1500 <600-1500	10^{21}	$5 \cdot 10^{21}$
Neutron Multiplier	High (n,2n) cross-section Low parasitic absorption	Be BeO, Be_2C Pb	<600 <1500 <300	10^{21}	$5 \cdot 10^{21}$
Reflector	High scattering Cross-section Low Parasitic Absorption	C, steel	Coolant temp.	10^{20}	$5 \cdot 10^{20}$
Shield	High macroscopic Absorption cross-section High gamma-ray attenuation	B B_4C Pb	100-200	$5 \cdot 10^{18}$	$5 \cdot 10^{20}$
Electrical Insulator	High dielectric strength High temperature stability (blanket) Low temperature stability (magnets) Resistant to radiation damage	Al_2O_3 MgO Phenolics	500-800 -269	$1.4 \cdot 10^{21}$ $3 \cdot 10^{13}$	$7 \cdot 10^{21}$ $3 \cdot 10^{15}$
Magnets	High Hc, Tc, Jc (S/C[c]) Low resistivity (non S/C) Strength at low temperature Ductile at low temperature	NbTi, Nb_3Sn, Va_3Ga(S/C) Cu, Al	-269	$3 \cdot 10^{13}$	$3 \cdot 10^{15}$
Thermal Insulator (low temp.)	Low thermal conductivity Good radiation damage resistance	Mylar	-269 to RT	$3 \cdot 10^{13}$	$3 \cdot 10^{15}$
Optical Materials (lasers)	High thermal stability Good reflectivity (Mirrors) Resistance to radiation damage	CdSe ZrO_2, TiO_2 Al, Au, Cu	100-200	Depends on design	

[a] Refers to all materials considered for this function;

[b] 100% plant factor

[c] S/C = superconducting.

2.2 Operating Environment for Fusion Reactor Materials

There are at least four unique features about the irradiation environment in a fusion reactor compared to a fission reactor. These are:

- neutron spectrum;
- charged particle flux;
- rate at which damage is generated; and
- photon flux.

The first three features are the most important with respect to materials performance, and we will now point out the major consequences of these effects. The reader is referred to the literature for a more detailed discussion.

2.2.1 Spectrum Effects

The increased energy of the D-T neutron (14.1 MeV) over those typical of a fission spectrum makes the quotation of simple neutron fluxes in Table IX-XI somewhat misleading, because the displacement rates are different and there are many more potential nuclear reactions to contend with (for example (n,n'p), (n,n'α), and (n,2n)). An appreciation of the spectral differences can be gained from Figure IX-8, where the neutron spectra from a fission reactor [IX-72], a typical fusion reactor [IX-73], a D-T neutron source [IX-74], and a D-Li stripping source [IX-75] are given. (The latter spectra are important for test facilities.) These numbers are plotted on an absolute scale so as to reflect the flux level as well as the energy spread of neutrons in these systems. The fusion spectrum has the traditional peak at 14 MeV, followed by a down-scattered spectrum that peaks over several hundred keV. This is contrasted to the fission spectrum, where the neutrons are emitted with a mean energy of approximately one MeV. Current D-T neutron sources are unable to provide sufficient backscattered neutrons to cause a significant deviation from the monoenergetic source, while stripping sources such as D-Be or D-Li provide a broad range of energies which depend on the incident deuterium energy. For a 33 MeV deuterium ion on Be the neutron energy varies from a maximum of about 33 MeV to below 1 MeV with a maximum in flux at about 18 MeV.

The importance of such spectral effects on one particular nuclear reaction is given in Table IX-XII. There the spectral averaged cross-section for the helium gas reactions in metals are listed for a fusion reactor and a light water fission reactor. This table reveals that the range of average spectral (n,α) cross-section for the metals examined here is 100 to 1000 times higher in fusion reactors than in fission systems. (This is reversed for Ni containing alloys for reasons explained elsewhere [IX-77, IX-78].) The ratio is even higher

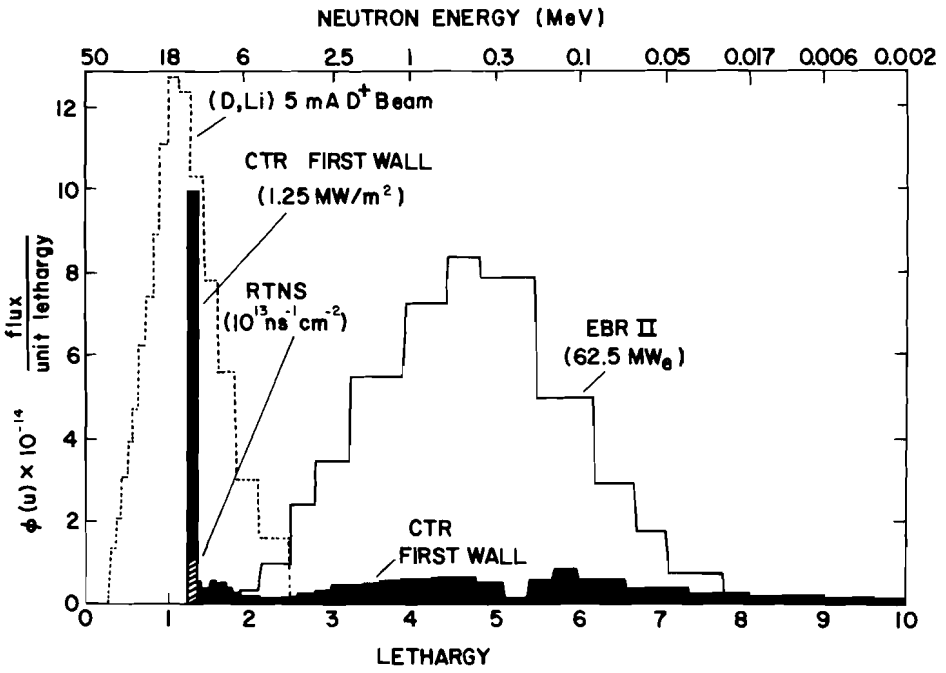

Figure IX-8: Typical Neutron Spectra for Various Nuclear Facilities

for the (n,2n) reaction in all metals. Such large differences are partially compensated for by a flux in fission reactors that is an order of magnitude higher, but taking both flux and spectrum into account we find that the magnitude of gas generation is still *much* higher in fusion reactors (see Table IX-XIII). This is also true for many other transmutations to non-gaseous isotopes.

2.2.2 Charged Particles

Fuel particles leaking from the plasma, as well as reaction products such as helium, will strike the first solid surface with considerable energy. The D and T energies will range from around 0.1 to 20 keV, while the helium energy can be as high as 3.5 MeV. Normally, most of the helium will be thermalized in a magnetically confined plasma so that the average energy will be about 100 keV, or even less. However, in laser systems, there may be quite high energy ions from the pellet, up to hundreds of keV with Z equaling 6 to 60. The particle fluxes vary not only with the reactor concept but also within a given design. For example, whether or not a TOKAMAK has a divertor

Table IX-XII: Effect of Neutron Spectrum on Helium Production Cross-section--Spectral Averaged Cross-section (mb)

Element	LWR[a]	CTR[b]
Mo	0.046	4.53
Nb	0.026	2.37
V	0.06	5.24
SS 316	≈60[c]	20.4
Al	0.28	32.5

[a] High flux isotope reactor (HFIR), core center [IX-76];

[b] UWMAK-I [IX-73], first wall;

[c] Due to thermal (n,α) reaction in Ni59 [IX-77], the value given here is valid after one year of exposure.

Table IX-XIII: Helium Production Rates in Various Potential CTR Materials (appm/yr[a])

	Fusion[b]	Fission[c]
Mo	47	2
Nb	24	1
V	57	0.3
Al	330	8
SS 316	210	5
C	3000	34

[a] 100% plant factor, see [IX-78];

[b] UWMAK-I spectrum [IX-73], 1 MW/m^2;

[c] EBR-II spectrum [IX-72].

can make a difference of one to three orders of magnitude in the particle flux. Similarly, if a Mirror reactor makes use of direct energy conversion, the charged-particle flux to various components can vary by several orders of magnitude. Typical particle fluxes in TOKAMAKs might vary from 10^{12} to 10^{15} p/cm^2·sec. Mirrors will have fluxes to the first walls on the upper side of this scale as will Theta Pinch machines. Fluxes from laser reactors will actually be even higher, if the results are normalized on the basis of 1 MW/m^2 of neutrons.

2.2.3 Burn Cycle and Neutron Flux Effects

It is difficult to make general statements about this kind of effect because there are several potential (at least five) avenues to fusion reactors, and they all represent drastically different burn cycles. We will try to put all of these cycles in perspective with respect to the following quantities (see Table IX-XIV):

- time over which neutrons are produced;
- time in which damage is done in materials; and
- time between burns.

The information in Table IX-XIV shows that the neutrons can be produced in burns which last from 10^{-9} to 10^{6} sec separated by times of 0.01 to 10^{5} sec. For all systems in which the burn time exceeds the neutron slowing-down time (approximately a μsec), the damage rate is constant over the burn time. However, in electron beam or laser systems, the neutrons are produced in times much shorter than the slowing-down time, and consequently the damage occurs at a relatively long time after the initial burst of neutrons.

In principle, the Mirror system could run in a steady state so that a precise assessment of burn dynamics is not meaningful. However, it is unlikely that any system as complex as a fusion reactor could run continuously for more than a month without a mechanical failure which would require a shutdown of the reactor. Therefore, we have somewhat arbitrarily chosen $3 \cdot 10^{6}$ sec as a typical operating time.

The situation for TOKAMAKs is somewhat unclear at the present time depending on the rate of buildup of impurities in the plasma, or the amount of flux swing that can be reasonably incorporated into the transformer coils. It is quite possible that if impurity confinement times are significantly longer than the confinement times of fuel atoms, the D-T burn cycle could be limited to as little as 100 sec in a reactor. On the other hand, if the impurities diffuse out of the plasma at a sufficiently rapid rate, then economic limitations of incorporating flux swings of more than 500 Vsec may limit the length of the burn to a few thousand sec before the magnets would need to be reset.

Theta Pinch and solenoid reactors limit their burn times to a few hundred msec due to a complex tradeoff between size of the plasma column, reasonable magnetic fields and rates of field buildup, and assumptions on burn dynamics. It is unlikely that the burn times would be greater than one sec, and values of less than 10 msec may be uneconomical in a magnetically confined system.

If one assumes that, nominally, a time-averaged 1 MW/m^2 wall loading is required for economical reactor operation, then one can obtain a rough approximation of the instantenous 14 MeV neutron flux (in $n/cm^2 \cdot sec$) to the first wall. This ranges from approximately $5 \cdot 10^{13}$ for Mirrors and TOKAMAKs, to approximately $4 \cdot 10^{14}$ n/cm sec for Theta Pinch and solenoid reactors, to as high as $4 \cdot 10^{17}$ to $4 \cdot 10^{19}$ $n/cm^2 \cdot sec$ for laser and electron beam reactors. Of course, if backscattered neutrons are included, these values would have to be increased by factors of roughly 5 to get total neutron fluxes. We have included typical values for fission reactors in Table IX-XIV, which include all the fast ($E > 0.1$ MeV) neutrons. Even if the TOKAMAK and Mirror numbers were adjusted to include backscattered neutrons, we would find that the total neutron flux is considerably lower than in a fast reactor. However, we shall see later that spectrum effects alters this picture in ways which are material dependent.

A somewhat imperfect but more reasonable way of comparing the potential damage rates in fission and fusion reactors is to calculate the theoretical fraction of atoms displaced per unit time of exposure to the irradiation environment. This unit, called the dpa for displacements per atom, does not include transmutation effects, or the amount of spontaneous and thermal recombination of the point defects. However, it does account for the probability that reactions will take place initially, and for the amount of energy which will eventually be transferred to the lattice atoms in nuclear encounters. Several authors have calculated displacement cross-sections [IX-79 to IX-82] and have noted that 14 MeV neutrons have dpa cross-sections which are about 4 times those of 1 MeV neutrons. They also find that the absolute magnitude of the dpa cross-section for heavy elements differ by less than 20 per cent. The situation for the low Z elements (for example C, or Al) is somewhat different, in that there is relatively little difference between 14 MeV and 1 MeV neutrons with respect to dpa values. This results from the predominance of low-angle scattering and the low threshold for ionization losses in these elements. Such an effect means that, neglecting the spatial distribution of defects on a microscopic scale and transmutations, 1 MeV neutrons are nearly as damaging as 14 MeV neutrons, and fission reactors probably make good simulation devices for displacement damage in low Z elements.

We can use the concept of displacement cross-section to calculate the damage rates in various fusion concepts. For example, we know that the dpa rates in TOKAMAKs are approximately

Table IX-XIV: Potential Burn Cycles for Various Fusion and Fission Reactor Concepts

Reactor Concept	Anticipated Neutron Pulse Length (sec)	Time of Damage per Pulse (sec)	Time Between Burns (sec)	Number of Cycles Per Year (80% plant factor)	Instantaneous 14 MeV Neutron Flux at an Average 1 MW/m^2 Wall Loading During Burn[b]
TOKAMAK	100-5000	100-5000	10-500	$5000-2 \cdot 10^5$	$\approx 5 \cdot 10^{13}$
Mirror	$3 \cdot 10^6$ [a]	$3 \cdot 10^6$	10^5	10	$\approx 5 \cdot 10^{13}$
Theta Pinch and Solenoids	0.3	0.3	3-10	10^6	$\approx 4 \cdot 10^{14}$
Laser	10^{-9}	10^{-6}	0.01-1	10^7-10^9	$\approx 4 \cdot 10^{17}$ to $4 \cdot 10^{19}$
E-Beam	10^{-9}	10^{-6}	0.01-1	10^7-10^9	$\approx 4 \cdot 10^{17}$ to $4 \cdot 10^{19}$
LWR-Fission Reactor	$3 \cdot 10^6$ [a]	$3 \cdot 10^6$	$\approx 10^5$	≈ 10	$\approx 2 \cdot 10^{14}$ [c]
LMFBR-Fission Reactor	$3 \cdot 10^6$ [a]	$3 \cdot 10^6$	$\approx 10^5$	≈ 10	$\approx 2 \cdot 10^{15}$ [c]

[a] Limited by mechanical failures rather than physics considerations;

[b] Units of $n/cm^2 \cdot sec$, backscattered neutrons would increase the numbers by factors of approximately 5;

[c] Maximum fluxes core centers, $E > 0.1$ MeV.

10^{-7} dpa/sec per MW/m^2, and roughly the same for Mirror reactors. However, the instantaneous dpa rates in a Theta Pinch are about 10 to 100 times higher, and those in a laser system are about 10^6 times higher. This situation (and the variation throughout a blanket model of approximately one meter thickness) are summarized in Figure IX-9 for SS 316. The displacement rates in the EBR-II reactor are also included in Figure IX-9, and the maximum is higher than all of the values for the TOKAMAK, but actually an order of magnitude lower than those in Theta Pinch first walls, and six orders of magnitude smaller than in laser (or electron beam) fusion reactor blankets.

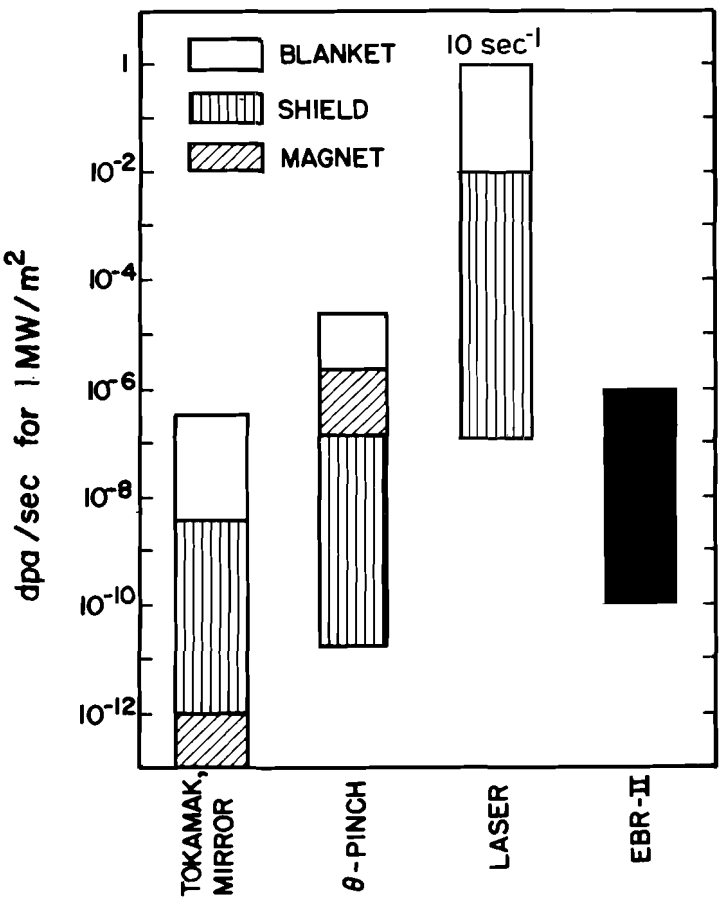

Figure IX-9: Instantaneous Neutron Displacement Rates in D-T CTR First Walls

The effect of damage rates on physical processes such as void formation, creep rates, fatigue, etc. is not very well known at the present time. It can be dangerous to think that one is simulating pulsed damage in a fission reactor, which can in fact produce the desired total dpa levels but only in a steady-state fashion. We should be sensitive not only to the fact that some instantaneous dpa rates in fusion reactors are 10^5 to 10^6 *higher* than in fission reactors, but also that some are actually a factor of 10 or more *lower* (i.e. TOKAMAK blankets, shields, and magnets).

Recent theoretical work [IX-83 to IX-85] shows that not only does the rate of producing damage alter such phenomena as the nucleation and growth of voids, but these phenomena are also quite sensitive to the downtime (annealing time) between pulses. At the present time, there are no acceptable facilities in the world to test even the simplest of the rate theories, let alone try to incorporate the effects of the thermally induced stresses. The laser and electron beam reactor concepts are the most vulnerable in this respect, and entirely new radiation damage theories need to be developed for these concepts.

2.3 Effects of Fusion Reactor Environment on the Properties of Materials

It will not be possible to discuss in this subsection all of the problems related to bulk radiation damage that have been identified thus far. However, we will attempt to mention what we think are the most important problems, and try to show how they may affect the normal operation of a fusion plant. It is convenient to discuss them in three separate groups: dimensional stability, mechanical properties, and physical properties.

2.3.1 Dimensional Stability

As with most complex devices, close tolerances and high quality assurance will be required to assemble a fusion reactor. Once in operation, it will be important that these dimensions are closely maintained for vacuum tightness and that unreasonable stresses are prevented. Because of the sheer size of fusion devices (i.e. one to two square meters of internal surface area for every MW(e) generated), even small percentage changes can result in large dimensional variations. In the UWMAK series of TOKAMAK reactors, one finds that a 0.1 per cent dimension change on the outer blanket structure can result in a 10 cm change in circumference. While the structure would have to be built to accommodate such strains (which might easily be imposed by thermal expansion), it is obvious that additional expansions or contractions, which may be a function of time, will be extremely difficult to predict, accommodate, and control.

(a) Swelling Due to Voids

There is one major dimensional instability associated with metals when they are irradiated at temperatures of approximately 25 to 55 per cent of their melting point. The generation of vacancies at temperatures above which they are mobile, and the preferential absorption of the associated interstitials at dislocations, produce a situation where the vacancies become highly supersaturated and tend to precipitate into voids. The metals then decrease in density with the net result that significant swelling can occur. Values up to 120 per cent have been reported for steels. This phenomena is rather general as shown in Table IX-XV, where we list some of the materials in which voids have been observed. Unfortunately, Table IX-XV includes all the potential CTR materials proposed for fusion applications so that one must prudently plan on some limited swelling if the irradiation temperature is high enough and if the damage level exceeds a few dpa. The exact magnitude of swelling to be expected may be found elsewhere for fission-neutron studies [IX-86, IX-87]. The basic questions for fusion-reactor designers with regard to voids in metals are the following:

- What level of *uniform* swelling can be tolerated without compromising the vacuum integrity, or causing the flow of coolant to be reduced?

Table IX-XV: Metals and Alloys in Which Neutron Produced Voids Have Been Observed After High Temperature Irradiation

Pure Metals	Alloys
Al	2024-Al, 6061-Al
Cu	
Fe	SS 304, 316, 321, 347, 348
Ni	NiAl, Ni-Cu, Incoloy, Inconel
V	V-Ti, V-Cr-Ti
Nb	Nb-1Zr
Mo	TZM, Mo-0.5Ti
Ta	
W	
Pt	
Co	
Mg	

- What level of *non-uniform* (remember the dpa gradients) swelling can be tolerated in a fusion blanket without compromising the safe operation of that reactor?

- What effect will high helium generation rates have on the data already obtained from fission-reactor studies?

- What effect will lower (TOKAMAK and Mirror reactors) or higher (Theta Pinch, electron-beam, or laser reactors) dpa rates have on the nucleation and growth or voids (compared to fission reactors)?

- What effect will periodic "anneals" between burns have on the resulting microstructure?

- What effect will the solid transmutation products have in the formation of voids?

- What effect will stress and/or the cyclic application of stress have on the resulting propensity to form voids?

None of the above questions have been satisfactorily answered, or even addressed in some cases. Such gaps in our knowledge will be very costly and time consuming to fill.

(b) Swelling Due to Gas Bubbles

The generation of insoluble gases (in this particular case, He) inside metals at high temperatures has been known to promote bubble formation and dimensional changes. This is not too serious in most metals (except for perhaps steels and Al) because the amount of gas generated is relatively low. On the other hand, there are certain materials which have been proposed for non-structural applications in fusion devices which could have serious problems with bubbles. Some of these include, Be, B_4C, C, and Li compounds such as Li_2O, $LiAlO_2$, Li_7Pb_2, or $LiSiO_2$ for example. Table IX-XVI lists the helium generation rate in these materials at different positions in a typical fusion blanket. The important points to note are the very high helium generation rates, several thousand to >15,000 appm per year in $LiAlO_2$. The B_4C is unlikely to be that close to the first wall except in special "burner" designs, so that values at approximately 100 cm are more appropriate. Even at that spacing, several thousand ppm of helium would be generated per year of operation. The effect of such high helium contents on the dimensional stability can be estimated as a function of bubble size and temperature from the following expression :

$$\frac{\Delta V}{V_o} = 100 \cdot N \cdot \left[\frac{rkT}{2\gamma} + b\right] \%\ ,$$

Table IX-XVI: Summary of Helium Production Rates in Non-structural CTR Materials (for 1 MW/m^2 wall loading)

Material	appm/yr[a]
Be	3,050
C	2,760
B_4C	3,600[b]
$LiAlO_2$	15,500

[a] Use UWMAK-III reactor, 100% plant factor;
[b] In shield 100 cm from first wall.

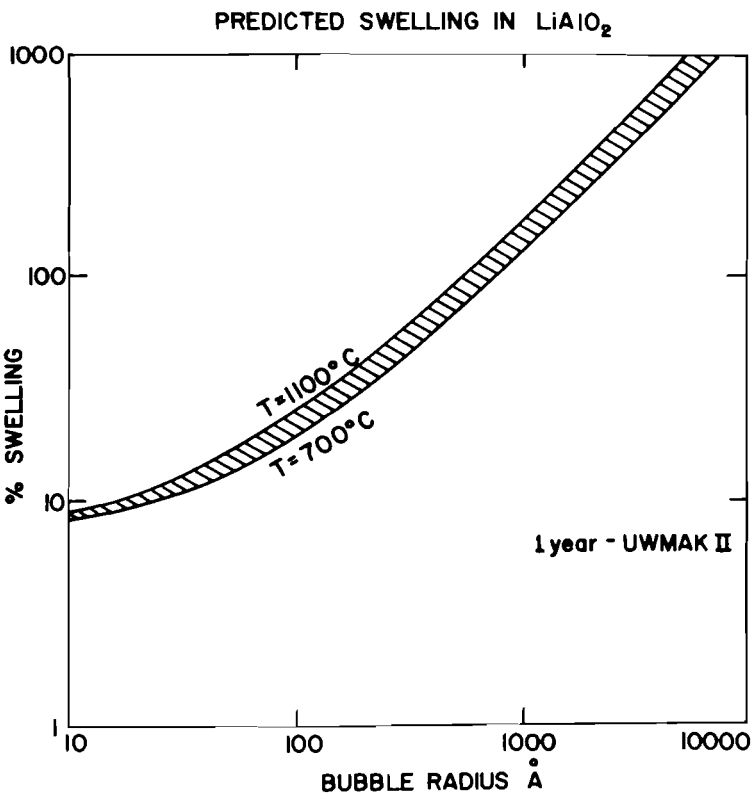

Figure IX-10: Calculated Effect of Bubble Size, Temperature, and Irradiation Time on the Helium Gas Induced Swelling in $LiAlO_2$

where

N = number of gas atoms per cm^3;

r = bubble radius;

T = temperature;

k = Boltzmann constant;

γ = surface energy;

b = Van der Waals constant.

One can get an idea of how serious this problem might be by calculating the swelling in $LiAlO_2$ after one year of UWMAK-II exposure. Figure IX-10 shows that even if the gas atoms collect relatively few vacancies, swelling values of around 10 per cent might be the characteristic after one year. Since $LiAlO_2$ is not a structural component but rather contained in cans, a 10 per cent volume change is probably not serious, but 50 per cent may be hard to incorporate into an economical design. In any case, swelling of non-structural materials could be an economic problem in that the reactor would have to be shut down to periodically replace swollen components.

(c) Growth

With the increased use of carbon in D-T fusion reactors for *(1)* impurity control [IX-65], *(2)* radiation damage reduction [IX-88 to IX-90], and *(3)* neutron reflection, it is important to understand the nature of the irradiation-induced growth mechanism in that material. There have been several reviews on the effects of *fission* neutron irradiation on the dimensional stability of graphite [IX-91, IX-92], and even a few assessments of how these data might be translated to fusion reactors [IX-93 to IX-95]. In general, neutron irradiation of carbon at elevated temperatures initially causes some shrinkage followed by expansion which eventually approaches a "run away" rate. Some typical data on nuclear grade graphite are shown in Figure IX-11. For purposes of calibrations, $1.4 \cdot 10^{21}$ n/cm^2 (fission) is equal to 1 dpa. This figure shows that useful lifetimes are typically 10 to 20 dpa at high temperatures (1000 to 1400°C).

Figure IX-12 shows the present status of experimental data from fission reactors. Also included in that figure are damage-temperature regimes that might be required for reflectors, plasma shields, or neutron spectral shifters. Note that current fission data (available from fission-reactor graphite) are sufficient to almost cover the needs of the reflectors. However, only limited data are available for 1200 to 1400°C carbon curtain concepts (roughly two years of equivalent dpa levels),

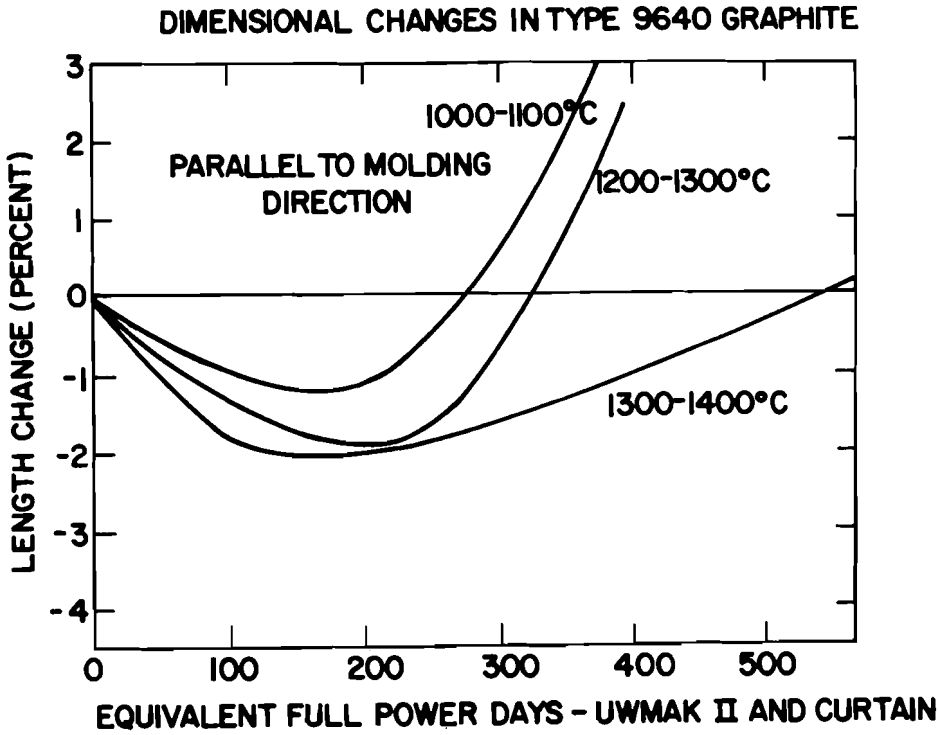

Figure IX-11: Effect of Molding Direction on Length Changes of 9640 Type Graphite [IX-95]

and there are no data available for the very high temperature ISSEC concepts [IX-88 to IX-90]. Such information must be generated before these ideas can be implemented in real reactor designs. Intuitively, one might think that as the irradiation temperature is raised above 1300°C the increased annealing would reduce the residual damage. However, a recent paper by M. Van Den Berg et al. [IX-96] suggests that such a trend may not be correct for all graphites, and in fact they find increasing damage rates up to 1400°C in some special grades. These results are at odds with the data in Figure IX-11 and the bulk of previous studies on graphite. Therefore, careful research is needed to understand this discrepancy.

It should also be stressed that the forms of carbon used for fusion reactors may be considerably different from those tested for fission reactor applications. Carbon cloths [IX-65], three-dimensional weaves [IX-65], and solid carbon walls [IX-88 to IX-90, IX-97] have all been proposed. The reactions of these forms of carbon to high-temperature neutron irradiation

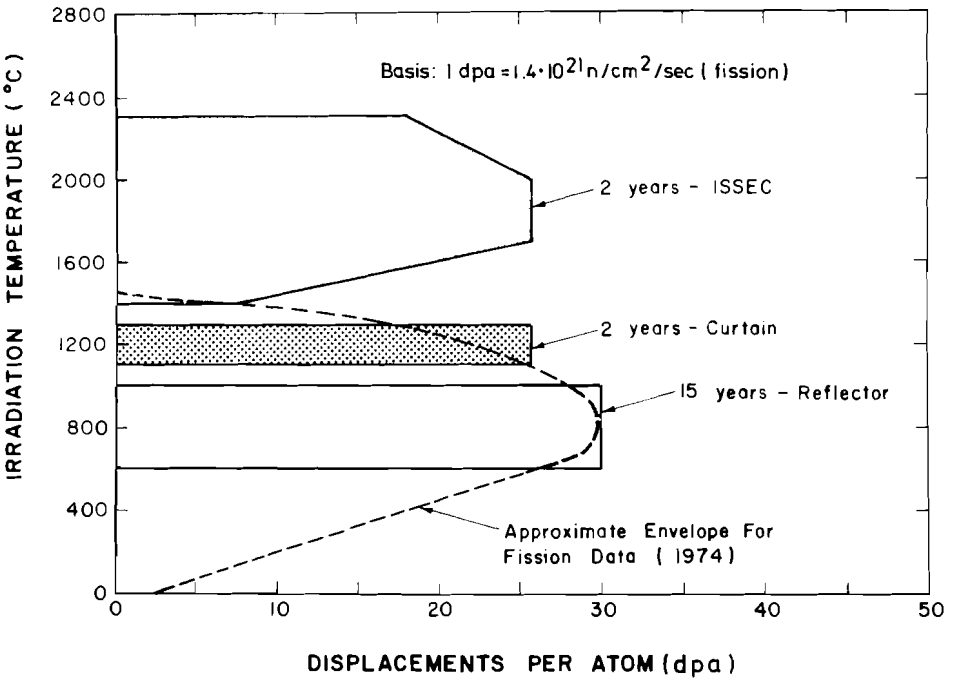

Figure IX-12: Comparison of Required Irradiation Data for Carbon in UWMAK-III to Data Available from Fission Reactors

may be considerably different from those for fuel-particle coatings (pyrocarbons), or forms extruded from anisotropic graphite. An entirely new irradiation program will be required to address these materials, and methods--which are largely unknown now--must be found to correctly simulate CTR conditions until suitable CTR neutron test facilities can be built.

2.3.2 Mechanical Property Changes That Could Be Important in CTR Materials

This is again one of those areas which is extremely difficult to summarize in the limited space available here. To be complete one should cover irradiation effects on such properties as :

- yield strength;
- ultimate strength;
- total elongation;

- uniform elongation;
- ductile to brittle transition temperatures;
- fracture toughness;
- creep;
- fatigue.

While all of these properties are important, we will try to briefly relate uniform elongation, creep, and fatigue to the performance of a fusion reactor.

(a) Ductility

It is absolutely essential that any massive structures such as fusion reactors have the ability to absorb a certain amount of strain energy without plastic yielding or fracturing. This will be required to offset thermal expansion between burn cycles, finite amounts of non-uniform swelling, or simple fabrication defects. The fact that the reactor will be extremely radioactive and therefore inaccessible except for remote techniques, coupled with the high cost of having an entire power plant off the line because of a single component failure, mean that the designers will need as big a "safety margin" as possible to keep the plant running. It is not easy to establish what that margin will be until a very detailed reactor design is available. However, we can take some lessons from the LMFBR program; there it is determined that the component must be changed when the properties of the fuel cladding are degraded such that a strain of more than 0.4 per cent exceeds the uniform elongation limit. It would be naive to simply assume that the same limit applies to perhaps a first wall of a fusion reactor, which must maintain absolute vacuum tightness over a 1000 m^2 in the face of changing magnetic fields, temperatures, flow rates, damage rates, and environments. The probabilities for failure are greater, and the time required to correct the fault will be longer in fusion reactors than those required to pull out a defected fuel element in a fission reactor. Intuitively, we would expect the design limit of a fusion reactor to be much more liberal than that of a fission reactor, allowing perhaps for a uniform elongation of as high as one per cent, but no one can say with certainty what it might be today.

There is only one metallic structural material for which we have enough data to estimate what neutron irradiation at elevated temperature might do to the uniform elongation. That material is SS 316. There are fast reactor data up to approximately 20 dpa (only a few appm He) at temperatures up to 650^{0}C, as plotted in Figure IX-13. It can be seen that operation up to roughly 20 dpa would result in uniform elongation (U.E.) values of approximately 0.5 per cent. These data (without the appropriate helium, however) also show that the one per cent limit would be reached in only a few years of 1 MW/m^2 exposure.

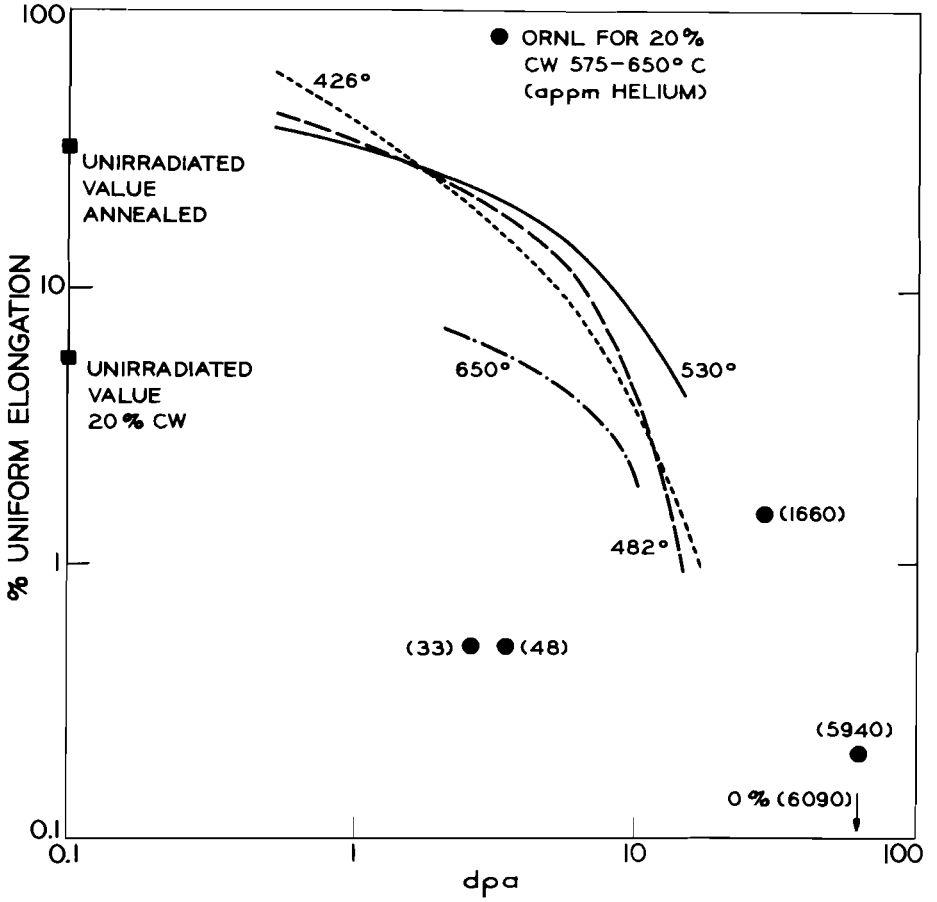

Figure IX-13: Effect of Neutron Damage on Uniform Elongation of SS 316

A very fine experiment has been conducted at ORNL to establish the effect of very high helium (several thousand appm), high dpa (up to roughly 90) and high temperatures (up to 650°C) on the uniform elongation of SS 316 [IX-98]. These results are also displayed in Figure IX-13. Unfortunately, it shows considerable scatter with some data points, predicting an 0.5 per cent ductility at He levels of <50 appm He at 575°C, and others showing the same or better ductility at roughly 90 dpa and 6000 appm He. Therefore, it is difficult to place a definitive wall life, unless one were to use the most pessimistic data. Such an approach would yield a life of two to three months in a reactor like UWMAK-II. If one uses the U.E. design limit of one per cent, the situation becomes much worse. In fact, it is

quite possible that the wall life would be less than two years even with the optimistic data. Above 650°C essentially no ductility remains after 90 dpa and 6000 appm.

The whole point of this exercise is to point out again that the high helium generation rate will probably place an upper temperature limit on the first wall life, regardless of the corrosion or creep behavior of the material. Secondly, it says that even for the only material on which we have data, the choice of design limit can only change an impossible situation (wall life is less than two months) into a difficult one (wall life is of only a few years), depending on the assumptions of tolerable ductility.

No such information on high helium contents exists for the other engineering materials (Al, Mo, Nb, V, etc.), because there is no corresponding quirk of nature such as the large thermal (n,α) cross-section for Ni59 in the other metals [IX-77]. Therefore, we must again come up with techniques that as to now are not known for testing these materials, to provide a backup for the only material on which we have some high-helium-content data. This is not a very comfortable position to be in, and it could require a great acceleration of the construction of D-T neutron-source facilities in order to solve the problem.

(b) Potential Creep Problems in D-T Fusion Reactors

As with any new energy source, fusion must demonstrate, among other things, that it can produce energy cheaper and with less environmental impact than fossil fuels and fission reactors. The desire for high efficiency normally means high temperatures, and each new design of a fusion reactor pushes its structural material to the stress limit. It is well known that the combination of high temperatures (close to half the melting point) and high stresses will cause materials to plastically deform over long periods of time. It has also been recently demonstrated that a superposition of neutron irradiation can increase the deformation (creep) rate over the thermal values [IX-99]. Hence, all three ingredients required for gross deformation are present in a fusion reactor blanket, and we should expect that creep-rupture lives of candidate materials will have to be further lowered over their unirradiated values.

E.E. Bloom and F.W. Wiffen [IX-98] have found that creep-rupture lives of SS 316 were reduced by 50 per cent after 575-625°C irradiation compared to their non-irradiated values, and there is no particular reason to expect that this would be different with the refractory metals. Therefore, if we want to have at least a two-year wall life (around 17,000 h) then stresses should be <10,000 psi in stainless steel. When appropriate safety factors are included (i.e. factors of roughly two) it is questionable whether a material like SS 316 can withstand the thermally induced stresses in the first walls.

Even if the first walls and coolant pipes did not rupture, a deformation of 0.5 per cent may significantly complicate maintainance procedures. For example, current reactor designs rely on periodic changing of the first walls due to radiation damage. This requires that modules can be easily removed and replaced remotely. A 0.5 per cent shape deformation (e.g. 5 mm in a panel of 1 m length) may cause first wall panels to "stick", or may make insertion of a new one an impossible job.

Since, at the present time, there are absolutely *no* irradiation-creep data for 14 MeV neutron bombardment of *any* material, one must ask the following questions and set up research programs to answer these questions:

- What will be acceptable creep levels in TOKAMAK, Mirror, Theta Pinch, electron-beam and laser reactors?

- What is the effect of the dpa rate (from approximately 10^{-7} per sec steady state to approximately $10^{-5}$and one per sec instantaneously in pulsed systems) on the thermal creep rate in potential CTR metals and alloys?

- Will the high helium generation rate associated with fusion significantly reduce the creep rate in metals?

- What effect will solid transmutation products have on creep rates?

- How much of a safety factor ought one apply to creep-rupture lives (once they are determined) for fusion reactors where downtimes could be much longer to replace failed components, and more expensive than in fission reactors?

One last comment on the generation of data to answer the above questions. It is relatively worthless to spend a great deal of money on post-irradiation creep studies. Of all the critical mechanical properties, this one should be measured in-situ. Unfortunately, there are very few fission reactors where even one position in the core is instrumented to perform such tests. The costs of capsule design and associated equipment are also quite expensive, which makes the cost per data point truly enormous. A successful irradiation creep study program, first of all, needs a realistic neutron source (there are none at this writing except for perhaps thermal neutron reactors for Ni containing alloys); secondly, large sums of research money (a million dollars for a capsule associated equipment, and personnel for a few months of testing of one material is not unreasonable); and thirdly, years of time are required to cover all the experimental conditions and materials. Such a program has not even begun as of 1976 and may represent a severe bottleneck to high power reactors (for example, FERF, or EPR) operation during the next decade (see Chapter III).

(c) Fatigue, Perhaps the Achilles Heel of Pulsed Fusion Reactor Concepts

Fatigue, like creep, is recognized by everyone as a potential problem for fusion reactors. Unfortunately, we know even less about the basic mechanisms of fatigue and the effect of irradiation on it than about creep, and there are even less data.

It is fairly clear where the fatigue problems stem from in TOKAMAKs (5,000 to 10,000 pulses per year), Theta Pinches (2 to 3 million pulses per year), or laser and electron beam reactors (30 to 300 million pulses per year). These stresses and strains are inherent in the plasma-physics of the concept, and only the Mirror has the potential for a relative steady-state operation. Unfortunately, the quantitative stress and strain cycles for these reactor concepts have not been clearly defined, so that a detailed analysis of this problem cannot be made today.

Finally, the data for fatigue lives should come from in-situ tests, or tests which closely resemble the operating conditions of particular reactor concepts. Such tests will again be costly, time-consuming, and difficult to simulate using non-fusion neutron sources. There are very few LMFBR or LWR data to build on here, in contrast to the case for creep, ductility, void swelling, growth, etc. Theoretical background is almost completely lacking and standards for conducting and assessing irradiation-fatigue tests are largely unknown. In short, there is still a long way to go in this area, and lack of success could prevent some fusion concepts from ever surpassing the proof of principle phase.

2.3.3 Some Physical Properties of CTR Materials That Depend on Radiation Damage

Nearly all of the physical (and thermal) properties of CTR materials will change somewhat because of 14 MeV neutron bombardment. However, only a few of them have been identified as significant (perhaps because only a few have been investigated with fission neutrons, let alone 14 MeV neutrons). We will make only a few comments here and fully expect that research in the next few years will uncover new problems, and perhaps some solutions.

(a) Electrical Resistivity

This property is mainly important for insulators and only of marginal importance for metals. A comprehensive review of the state-of-the-art for insulators has been recently released [IX-100]; it has been concluded that *(1)* there is a general lack of data on in-situ resistivity changes for fission neutron

bombardment, and a complete lack of data for 14 MeV neutrons; *(2)* isotropic crystal structures seem to be less susceptible to property degradation than highly anisotropic structures; *(3)* rate effects have not been established;and *(4)* no information is available on the effects of high helium contents or on the consequences of generating significant solid transmutation products.

Electrical insulators are absolutely necessary for Theta Pinch reactors to prevent excessive power loss in the first walls. Mirrors and TOKAMAKs will also require insulators for neutral-beam injectors or pellet injectors. It is not clear how much of a neutron exposure these insulators will experience, because there may be a possibility of some shielding or placing line of sight insulators far back into the blanket, where they would intercept a relatively small solid angle. There may be another insulator requirement for TOKAMAKs if they use radio frequency (RF) heating. Filling the waveguides with dielectrics can significantly reduce their size, but such effects as high temperature gradients in thick insulator blocks remain to be investigated [IX-101].

The field of irradiation effects on dielectrics by high energy neutrons is not very well established or coordinated, certainly not at the level required for full fusion-reactor development. Theories are essentially non-existent for the effects of helium on the dielectric strength. (This is important because most insulators contain oxygen which has a high (n,α) cross-section.) Lack of appropriate neutron sources and in-situ facilities greatly hamper a successful program in this area.

The electrical resistivity of metals is of interest as it would not be desirable to have large power dissipations in the flux shields for toroidal field magnets in TOKAMAKs during the burn cycle. This is also true for the walls of waveguides in RF cavities. Few high temperature-high fluence resistivity data are available from fission facilities, and again none from higher neutron facilities. J. Moteff et al. [IX-102] have measured the post-radiation resistivity increase in Mo irradiated to 10^{22} n/cm^2 at temperatures from 400 to 1200^{0}C. It was found that at that exposure level, the irradiation-induced resistivity increase was <1 micro-ohm•cm, which is <3 per cent of the electrical resistance due to thermal vibrations at 1000^{0}C. Hence, it appears that the production of voids and dislocation loops at these exposures does not cause an unmanageable resistance increase.

One word of caution before we leave this area: The electrical resistivity of metals at high temperature should be subject to transmutations, and these are not adequately simulated by fission neutrons. Doping studies (in the absence of irradiation) may help to understand these effects.

(b) Radiation Damage to Superconducting Magnet Materials

This problem, which is peculiar to fusion, luckily is solvable by increased shielding in the case of TOKAMAKs and Mirrors. Of course, this means higher capital costs, and adversely affects the fusion-power economy. Hence, a relatively straightforward compromise between damage to magnets and cost of increased shielding and larger magnets will have to be made in these reactors.

The radiation damage susceptibility of at least five materials will have to be examined for superconducting magnets as they are now envisioned:

- superinsulation (e.g. mylar);
- structural material (e.g. austenitic steel or Al alloys);
- stabilizer (e.g. Cu, or Al);
- superconductor (e.g. NbTi, or Nb_3Sn);
- electrical insulator (e.g. epoxy).

Previous analysis of these problems reveals that the superinsulation and stabilizer are the most sensitive to radiation effects; A15 compounds like Nb_3Sn follow closely behind. NbTi has a rather good resistance to property degradation, as will be shown later.

The problem with organics such as mylar is that they become brittle and crumble. They could lose the ability to uniformly cover the cold magnets, and hence lead to larger refrigeration losses. Thresholds for observable effects are in the 10^7 rad range, and a 25 per cent reduction in ductility occurs at 10^8 rad [IX-103]. A recent analysis [IX-101] of a TOKAMAK reactor shows that a 1.5 m blanket "leaks" 10^6 rad per year, obviously leaving enough lifetime for even the most pessimistic designer. However, leakage of neutrons down beam or fueling ports, or out of divertor slots may cause local problems.

The irradiation of pure metals at liquid helium temperatures, for some time has been known to cause an increase in electrical resistance of these metals. Since the main function of a stabilizer in a magnet is to temporarily carry the current without significant heating in the event that a superconducting element goes normal, the increased resistance counteracts that objective. The rates of resistivity increase for pure Al and pure Cu have been determined by J.A. Horak and T.H. Blewitt [IX-104] and are plotted in Figure IX-14. Note that it requires approximately 10^{-4} to 10^{-5} dpa before the radiation damage resistance is of the same order of magnitude as the residual resistance due to impurities, imperfections, and lattice vibrations at 4.2 K. Somewhat arbitrary design considerations might state that one should remove the damage (by annealing at a higher temperature) when the irradiation induced resistance exceeds the residual resistance by 10 per cent. This time period

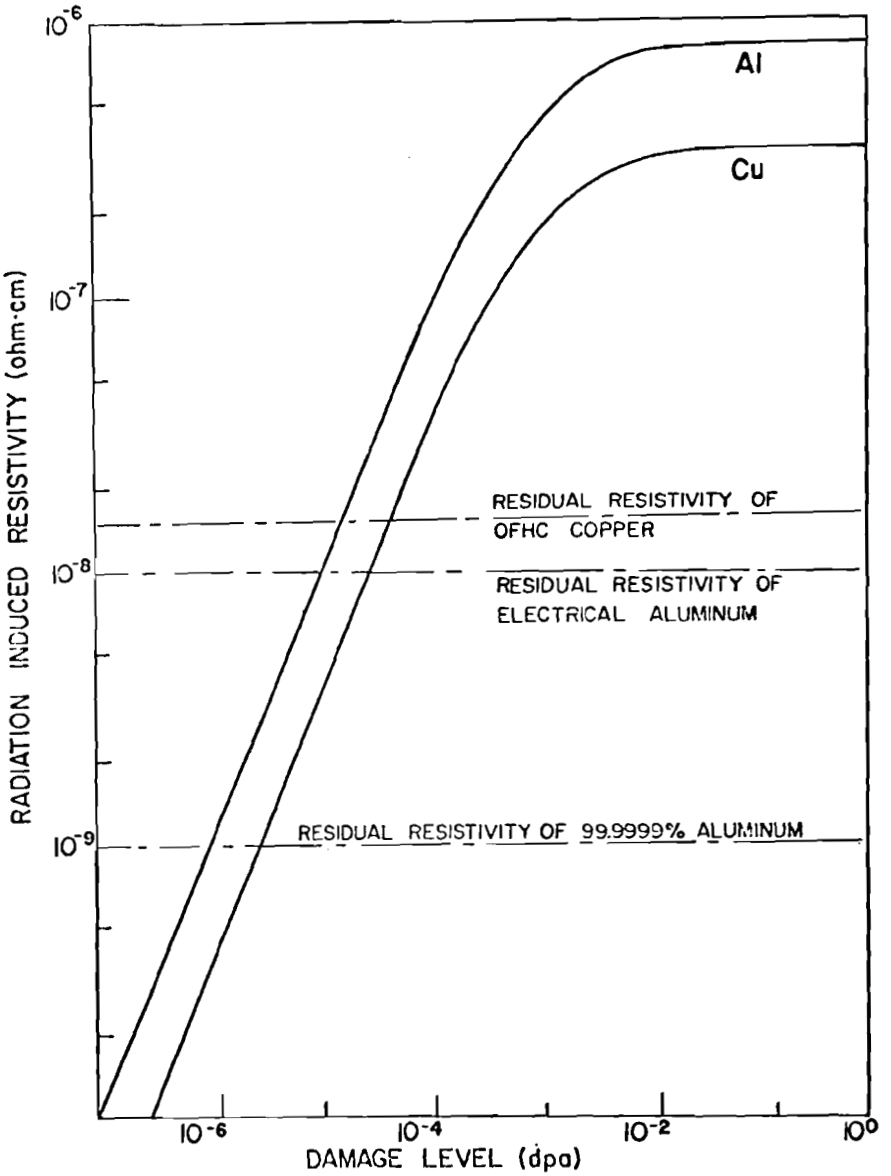

Figure IX-14: Radiation Induced Resistivity of Copper and Aluminum

can range from 1 to 60 years in various reactor designs.

The next area to consider is the effect of neutron irradiation on the critical properties of superconductors. There are usually two types of data that are reported in this regard: *(1)* samples which have been irradiated at room temperature (or above) and then tested at liquid-helium temperatures outside the reactor afterwards; and *(2)* samples which have been irradiated at liquid-helium temperatures and tested at the same temperatures without intermittent warm-up to room temperature. Unfortunately, there are very few of the latter data, and those of case *(1)* are not always representative of the true damage state. Not only are there fewer defects that remain after the higher temperature irradiation, but the increased mobility at higher temperature will cause the defects to form clusters or loops which might not be present in the "real" case of irradiation at liquid-helium temperature.

Two properties are of prime importance for superconductor in CTR magnets; these are the critical temperature (T_C), and the critical current density (J_C). The effects of fission

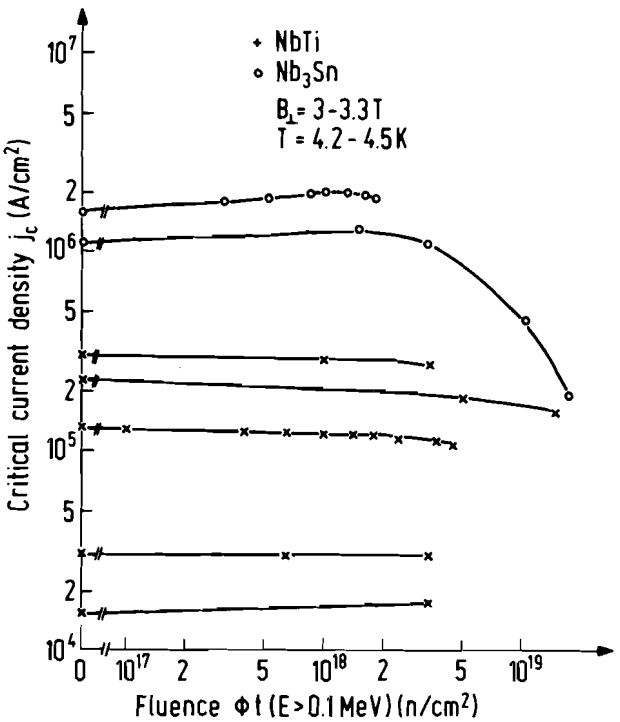

Figure IX-15: Effect of Neutron Irradiation on the Critical Current in NbTi and Nb_3Sn

neutron irradiation on the J_c of NbTi and Nb_3Sn are shown as a function of displacement damage in Figure IX-15 [IX-105]. (For purposes of comparison, 10^{19} n/cm^2 in fission equals roughly 0.004 dpa.) Considering the typical dpa rates (approximately adjusted for different atomic weights), one concludes that the J_c is changed by less than 10 per cent for both alloys in typical fusion environments (approximately 10^{-6} to 10^{-5} dpa).

The effect of irradiation on the T_c of several alloys and compounds has been studied by A.R. Sweedler et al. [IX-106] and is given in Figure IX-16. For practically all the A15 compounds, a significant drop in the T_c occurs at 10^{-3} dpa. Since NbTi is much more resistant to such degradation, it is not expected to pose a significant problem in fusion reactors.

In summary, appropriate blanket and shield design can reduce and even eliminate radiation damage which was a major problem in CTR superconducting magnets. However, the price paid is the extra cost of materials and the larger magnet design. A special effort must be made to verify these tradeoffs in integral tests at liquid helium temperatures.

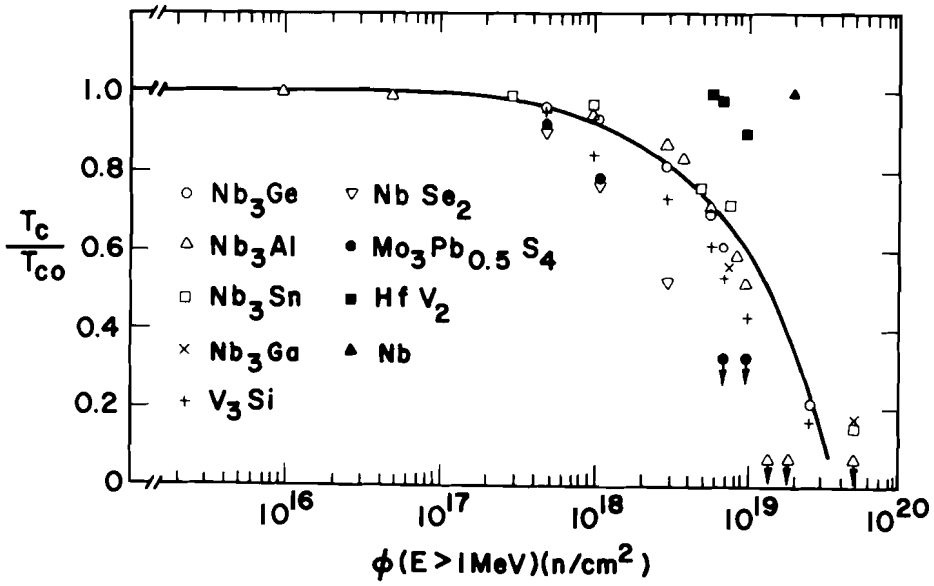

Figure IX-16: Effect of Ambient Temperature Irradiation on the Critical Temperature of Various A15 compounds

2.4 Discussion of the Importance of Neutron Radiation Damage on Commercial CTR Power Plants

The degradation of materials properties by neutrons results in at least the six following major effects:

(1) reduced efficiency;

(2) reduced plant factors;

(3) increased capital costs;

(4) increased operating costs;

(5) increases in the volume of radioactive waste which must be processed and stored; and

(6) demand on scarce elements.

(1) Reduced efficiency. The generation of helium gas tends to reduce the maximum temperature at which structural, breeder, and neutron multiplier materials of CTRs can operate for long periods of time. This in turn reduces allowable coolant temperatures, which in turn will lower the overall plant efficiency.

(2) Reduced plant factors. The fact that certain components of the reactor will have to be replaced before the full lifetime of the plant is reached, means that costly shut-downs must occur. The exact downtime is a function of many complex considerations, but some perspective on the costs can be obtained if one remembers that the revenue from a 2000 MW(e) plant is approximately $ 1,000,000 per day at 20 mill/kWh(e). Estimates for some reactor designs predict that approximately 30 days per year may be lost due to radiation damage, and changing the first walls costs approximately 30 million dollars per year per 2000 MW(e) plant in downtime alone [IX-107].

(3) Increased capital costs. Spare modules must be purchased at the start of the plant to replace those involved in the first change-out (thereafter, the costs are included in operating costs). Increased remote-handling equipment will be necessary to minimize the time involved in plant shut-dawn. Added hot-cell facilities may also be required. Shielding requirements for gamma rays emitted from damaged components (or good ones, for that matter) will also increase the overall plant costs. Waste storage facilities will have to be expanded beyond those required for components which fail for "conventional" reasons such as corrosion, machining faults, etc.

(4) Increased operating costs. Items *(1)*, *(2)*, and *(3)* combine with other costs to raise the cost of electricity as measured in mills per kWh. A rough idea of the sensitivity of this number to first-wall lifetime is shown in Figure IX-17 [IX-107]. This analysis, which is detailed elsewhere for UWMAK-I and II, reveals that if the first-wall lifetime is less than two years (at a nominal wall loading of 1.2 MW/m^2),

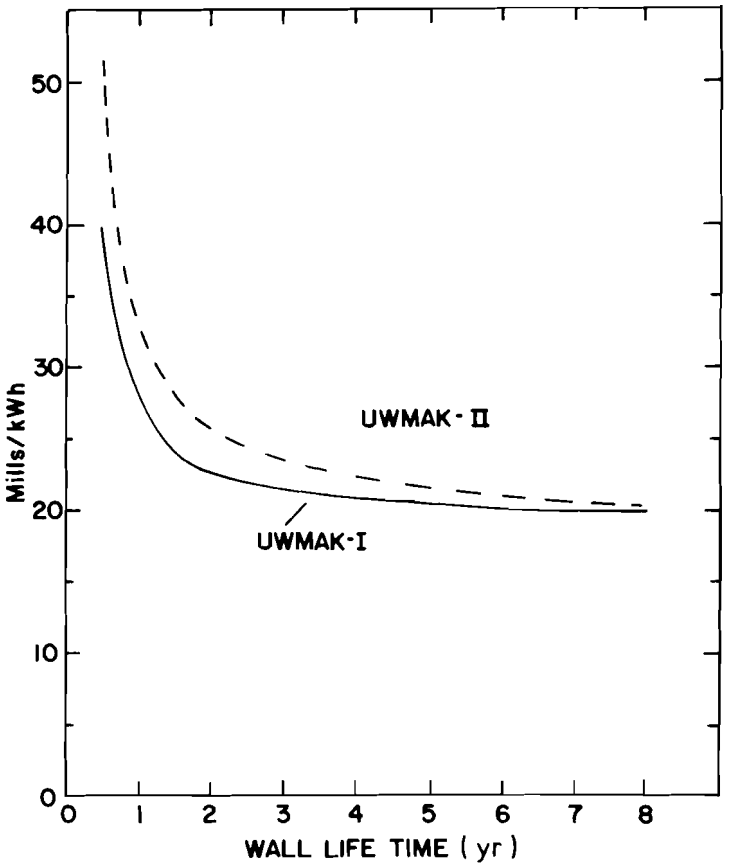

Figure IX-17: Effect of Wall Lifetime on Electricity Costs in UWMAK-I and II

the average cost of electricity rises dramatically. It also shows that the increased cost of lowering the wall life from 8 to 4 MW•yr/m^2 is only approximately 10 per cent of the total.

(5) Increases in the volume of radioactive waste which must be processed and stored. Most of the major reactor studies to date have made some assumptions about the first-wall lifetime. These are listed in Table IX-XVII, along with the metal system and the amount of material to be replaced per MW(e)·yr. This number is surprisingly constant considering the variation in design group, materials, and reactor power level. A reasonable average is approximately 0.4 t/MW(e)·yr. If we ever do get into a large-scale fusion reactor economy, such as 10^6 MW(e) by 2020 [IX-108], then this means that approximately 400,000 t of radioactive waste would be generated per year. Clearly such a number represents a potential problem in waste management.

Table IX-XVII: Summary of Radioactive Waste Amounts for Various CTR Reactor Designs

Reactor	System	Predicted Wall Life ($MW \cdot yr/m^2$)	Material Replacement ($t/MW(e) \cdot yr$)
UWMAK-I	SS 316	2.5	0.69
UWMAK-II	SS 316	2.3	0.49
UWMAK-III	TZM	3.4	0.31
ORNL	Nb-1Zr	>10	0.41
BNL	Al	3.8	0.27
LASL-ANL	Nb-1Zr	10	0.33

(6) Demand on scarce elements. When components become defective and radioactive at the same time, it is usually more economical to compact, process, and store them until the radioactivity decays to safe levels, than try to refabricate them. However, we see from Figure IX-18 that the decay times can take hundreds, if not thousands of years. Hence, for all intents and purposes, the replacement of these components will have to come from new elements. The disposal of perhaps 400,000 t/yr of SS 316 means that approximately 70,000 t of Cr must be supplied per year, along with appropriate amounts of Mn and Ni. In some cases, e.g. Be, there may be no choice but to reprocess the radioactive and contaminated metal, because world reserves are not adequate for a "throw away" economy.

Even if all the components had the same life as the reactor, there would be the problem of what one does with the radioactive structure when the plant becomes obsolete and a new one must be built. The blanket, shield, magnets, supports, and all equipment within three meters of solid material from the plasma will be too radioactive to dispose of in a conventional manner. These masses typically amount to approximately 50 t/MW(e), and will also place a severe strain on our limited resources as the second, third, fourth, etc. generation plants are phased out in the 21st century.

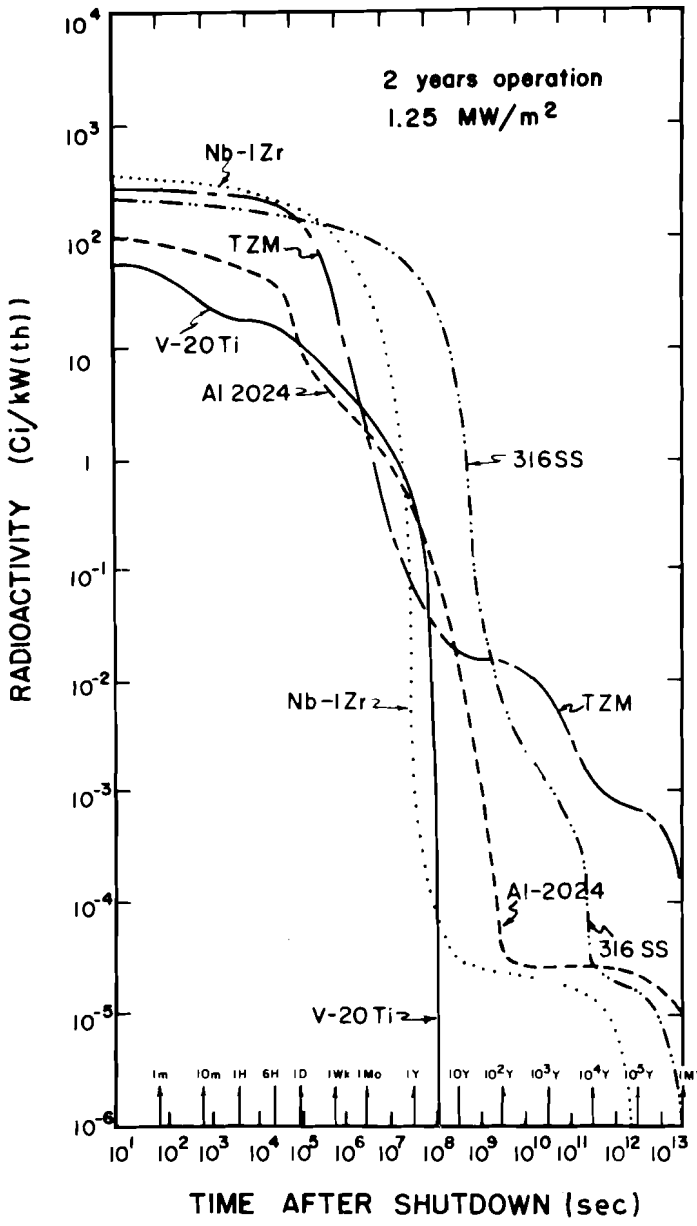

Figure IX-18: Radioactivity in Fusion Reactor First Walls After Shut-down

2.5 Summary

This has been a rather broad look at the neutron-damage problems currently envisaged for D-T reactors. Not all the problems have been discussed and, indeed, a whole class of conditions for fission-fusion concepts has been left out. However, it is hoped that the reader will begin to appreciate the concern of the materials-science community over the growing list of problems to be solved in the field of fusion energy. Undoubtedly more problems will be identified in the future. We must, therefore, reluctantly conclude that, next to the plasma physics problems, radiation damage is the second most serious obstactle to the commercialization of fusion power.

3. NORMALIZED MATERIALS CONSUMPTION

3.1 Introduction

The recent concern about the worldwide energy crisis, and the current optimism about demonstrating the technological feasibility of different energy systems, have prompted many scientists to examine the long-range effects of such energy systems. Such a long-range concern can be helpful in the early stage of development, because it may point out desirable (and undesirable) features of such an electrical economy, and give an estimate on the amount of materials to be made available in the next few decades.

While the choice of materials to be used in LMFBRs is rather definite now (except for modest improvements in future years), it is too early to make a definite choice on all the materials in the field of fusion research. But one must be sensitive to parameters such as thermal pollution (hence, the need for materials that can operate at high temperatures to increase efficiency); land despoilment (hence, the need for easily minable and readily available minerals); air pollution (relating to refining techniques); induced radioactivity, and the potential for its release; the demand to recource ratio (hence, the need for readily obtainable materials); local self-sufficiency (such that every country can have access to its supply of minerals); and finally cost.

It is the purpose of this subsection to present what we currently think will be the needs for materials in fission breeders and fusion reactors and then to compare these needs against the known reserves and resources. The reader must clearly recognize that, for fusion reactors, this assessment is only applicable to the state-of-the-art as of 1976, and that in five years the picture might have changed dramatically.

3.2 Fast Breeder Reactors

As of now, six prototype liquid-metal-cooled, fast-breeder reactors of the 250 to 350 MW(e) class are either under construction or already in operation. Detailed designs are available for the 600 MW(e) BN 600, and the 1200 MW(e) plants SUPERPHENIX and CFR, as well as conceptual design studies for plants in the power range of 1600 to 2000 MW(e). In addition, several conceptual design studies for gas-cooled fast-breeder reactors are known.

This gives a clear picture of the amount of materials needed for the construction and operation of fast breeders. Since all fast-breeder-reactor designs, known so far, are not dramatically different among themselves, the following presentation of the materials needs of liquid-metal-cooled fast-breeder reactors mainly uses data from SUPERPHENIX [IX-109] and SNR-2 [IX-110].

Materials requirements fall into two basic categories, the nuclear part and the conventional part of the plant.

For a liquid-metal-cooled fast-breeder reactor, these categories can be subdivided further as follows:

Nuclear part of the plant:

- core fuel (enriched PuO_2-UO_2 mixed oxide);
- blanket fuel (depleted UO_2);
- steel for fuel cladding and structural parts of the core;
- coolant (sodium);
- control material (boron carbide);
- reflector and shielding material;
- core support, and coolant vessel internals (steel);
- coolant vessel, and vessel cover or roof with rotating shield plugs for refueling;
- fuel transfer and handling machines;
- primary-coolant circuits including pumps, valves, tubes, and intermediate heat exchangers.

Conventional part of the plant:

- secondary-coolant circuits including pumps, valves, expansion tanks, and tubes;
- sodium-water steam generators;
- conventional water-steam part;
- turbines;
- generators.

Table IX-XVIII: Materials Consumption Rates for LMFBRs (data based on characteristic 1200 MW(e) LMFBR design of thermal output of 3000 MW(th), and yearly electrical output of $8 \cdot 10^9$ kWh(e))

Reactor Core Component	Inventory (kg)	Lifetime Until Discharge (yr)	Consumption Rate (kg/MW(e)·yr) PuO_2-UO_2	UO_2	B_4C	SS	Na
1. Reactor Core							
1.1 Core Fuel Elements							
Plutonium	4,600 initial core inventory (additional 150-300 kg/yr are produced by breeding						
Mixed-oxide fuel (17% enriched)	36,000	2.5	15				
Stainless steel	21,000	2.5				9	
1.2 Axial Blankets							
UO_2 fuel	22,000	2.5		9			
Stainless steel	12,000	2.5				5	
1.3 Stainless steel							
Upper axial shielding, fission-gas plenum, lower axial shielding, subassembly feet	97,000	2.5				40	
1.4 Radial Blankets							
UO_2 fuel	52,000	6-8		8			
Stainless steel	75,000	6-8				11	
1.5 Control and Shutdown Rods							
Control material B_4C	1,200	2.5			0.5		
Stainless steel	10,000	2.5				4	
Total Reactor Core			15	17	0.5	69	

2. Core Structure Plates							
Core support plates	110,000	30				17	
Core support structure	200,000						
Structure and shielding	180,000						
Core restraint system							
3. Reactor Vessel	1,100,000	30				40	
4. Rotating Plug System (including additional components)	800,000	30				28	
5. Upper Roof of Reactor Vessel in Case of Pool Type Reactors	900,000	30				30	
6. Primary Circuits (including tubes, expansion tanks, etc.) in Case of Loop Type Reactors	100,000	30				3.5	
7. Primary Pumps (without motor)	130,000	30				4.5	
8. Intermediate Heat Exchangers	425,000	30				15	
9. Transfer and Handling Machines	400,000	30				14	
10. Secondary Circuits (including tubes expansion tanks,etc.)	100,000	30				3.5	
11. Steam Generators	700,000	30				25	
12. Secondary Pumps	130,000	30				4.5	
13. Coolant							
Primary Sodium	3,350,000	30					115
Secondary Sodium	1,500,000						53
Total			15*	17	0.5	254	168

* These amounts of fuel must be reprocessed and refabricated. The real consumption rate of UO_2 fuel for fast breeders is roughly 1.5 kg/MW(e)·yr.

The conventional part of the plant is self-explanatory and therefore not discussed here.

In fast reactors, the PuO_2-UO_2 mixed oxide must operate between temperatures of several 100°C and 2800°C for about 2 to 2.5 years until roughly 10 to 15 per cent of the original fuel atoms are fissioned. The fuel cladding and core-structure steel operate between 150 and 650°C up to neutron doses of two to three times 10^{23} n/cm^2, or up to 150 displacements per atom (dpa). Core-support and coolant-circuit materials do not operate under such high neutron irradiation, but will have to resist thermal stresses, creep, and low-cycle fatigue loads.

In a typical 1200 MW(e) liquid-metal-cooled, fast-breeder reactor, the core fuel elements, together with their axial blanket zones and the absorber elements, are exchanged after 2 to 2.5 years. The radial breeder-blanket elements are discharged after six to eight years. All other components are designed to reamin in operation over 30 years, the full lifetime of the plant: the core and blanket-support structures, shielding structures, the reactor vessel with rotating shield plugs and refueling machines, the primary and secondary coolant circuits, etc. Under these assumptions, the material consumption rates as given in Table IX-XVIII must be considered.

There are five different categories of materials:

I - fuel which is recycled in the fuel cycle;

II - control and cladding materials which must be put to waste storage facilities;

III - structure materials which must be stored until their radioactive isotopes have decayed away and recycling can be considered;

IV - steel structural material which can be recycled after the entire plant has been decommissioned and decontaminated;

V - sodium coolant which could be recycled after appropriate clean-up and decay of around 30 years.

Table IX-XIX lists these categories of materials and their consumption rates.

Taking the data from Table IX-XVIII, and the composition of the various materials, one can list the consumption rates of non-fuel elements of the periodic table (see Table IX-XX). About one to two per cent of the plutonium is lost during fuel fabrication and reprocessing, and the rest is recycled by reprocessing. For control material it is assumed that boron carbide (B_4C) is used, whereas SS 316 and SS 304, respectively are used as cladding and structural steels.

Table IX-XIX: Categories of Materials Consumption

Classification	Reactor Material	Consumption rate (kg/MW(e)·yr)
Category I	Fissile and fertile Fuel	32
Category II	Control material	0.5
	Cladding material	69
Category III	Primary-circuit structural steel (cannot be recycled until after at least 100 years storage)	119
Category IV	Secondary-circuit structural steel (can be recycled after use)	66
Category V	Sodium coolant (can be recycled after clean-up and around 30 years of decay)	168

The reader should recognize that, while the numbers in Table IX-XX for the "nuclear island" are probably fairly accurate, those for the "balance of plant" (BOP) are certainly lower limits, because they do not include items such as turbine-generator, building materials, steel liners, etc.

It can be seen from the table that about six tons of non-fuel materials are required per MW(e) for the nuclear island, and at least two more tons per MW(e) can be readily identified with closely associated equipment. Roughly half of these amounts are due to sodium, the other half being due to steel components. Therefore, at least eight tons of critical elements are required per installed MW(e).

If the materials inventory and materials consumption figures are averaged over a 30 year lifetime (80 per cent plant factor), one arrives at a materials commitment of approximately 0.28 t/MW(e)·yr. Adding a few of the BOP values would raise this to roughly 0.4 t/MW(e)·yr. The significance of these numbers will be apparent when we consider the resource picture.

One may compare these materials needs to the world resources for different elements to determine the upper limit of energy MW(e)·yr which could be generated. Table IX-XXI compares world resources [IX-111, IX-112], and the materials needs listed in Table IX-XX, applying the same assumptions as used for fusion reactors in Sub-section IX.3.3.4.

Table IX-XX: Summary of Original Non-fuel Requirements for Fast Breeder Reactors

Element	Material Requirement[a] (t/MW(e))		Materials Commitment (t/MW(e)·yr)[d]	
	Nuclear Island[b]	BOP[c]	Nuclear Island[b]	BOP[c]
B	0.001	-	$4 \cdot 10^{-4}$	-
C	0.002	-	$1 \cdot 10^{-4}$	-
Cr	0.56	0.26	0.033	0.015
Fe	2.0	0.9	0.1	0.07
Mn	0.06	0.03	0.003	0.002
Mo	0.06	0.03	0.003	0.002
Ni	0.44	0.2	0.014	0.014
Na	2.8	1.3	0.12	0.06
Subtotal	5.9	2.7	0.28	0.16
U	0.09	-	0.004	-
Pu	0.004	-	NA	-
Subtotal	0.094	-	0.004	-
Total	6.0	2.7	0.28	0.16

[a] Total commitment of 30 year life;

[b] Everything inside and including pressure vessel;

[c] Balance of plant (including pumps, heat exchangers, etc.) but not buildings, turbines, generators, etc.;

[d] 80 per cent plant factor.

The amount of MW(e)·yr to be generated can be calculated by the following simple expression:

$$\frac{\text{(Resources Limit at Present Prices)}}{\left(\begin{matrix}\text{Initial Material}\\ \text{Charge in t/MW(e)}\end{matrix}\right\} \cdot \frac{1}{24\ \text{yr}}\right) + \left(\begin{matrix}\text{Makeup Material}\\ \text{in t/MW(e)}\cdot\text{yr}\end{matrix}\right)}$$

An 80 per cent plant factor is used, i.e. 30 years plant life correspond to 24 years of continuous operation.

Table IX-XXI: Summary of Maximum Possible Energy Generation and Materials Requirements for Fast Breeders Under Resource Limitations

Element	Maximum Requirement[a] (t/MW(e))	Materials Commitment (t/MW(e)·yr)	World Reserves[b] (10^6 t at present prices)	Maximum Energy Generated from Resource Limitations (MW(e)·yr)
B	0.001	$4 \cdot 10^{-4}$	20	$4 \cdot 10^{10}$
Cr	0.82	0.048	370	$4 \cdot 10^{9}$
Fe	2.9	0.17	276,000	10^{12}
Mn	0.09	0.005	72,000	$8 \cdot 10^{12}$
Mo	0.09	0.005	13	$1.5 \cdot 10^{9}$
Ni	0.64	0.028	24	$5 \cdot 10^{8}$
Na	4.1	-	large	large
Ti	0.0001	-	181	10^{13}
U	0.09	$1.5 \cdot 10^{-3}$	see Chapter II	$2 \cdot 10^{9}$

[a] Values for nuclear island and balance of plant excluding turbines, generators, and buildings;

[b] [IX-111, and IX-112] world estimates as of 1976.

If one were to account for the total balance of plant figures of the LMFBR plant, one would have to apply a factor of perhaps two to the figures in Table IX-XXI.

One can conclude from Table IX-XXI that with the exception of nickel, all elements lead to a higher potential energy generation than uranium. But only about ten per cent of the steel structure of the LMFBR plant is used within or in the vicinity of the core. Approximately 90 per cent of the steel would not become radioactive and could be recycled. The use of ferritic steels, which is possible for LMFBR plant components, would improve this picture even further.

For a more general discussion and a comparison with the materials requirements of fusion-reactor systems, see Subsection IX.3.3.4.

3.3 Fusion Reactors

3.3.1 Methodology

Even so there have been at least 21 large-scale reactor designs (see Chapter IV.2), attempts to tabulate the materials required have been made for only seven of them. The major feature of those reactors are given in Table IX-XXII. For more details, the reader is urged to consult the main reports from which these data were extracted.

The materials requirements fall into two basic divisions, nuclear island and balance of plant (BOP), which are then subdivided as follows:

Table IX-XXII: Major Features of CTR Reactors Which List Materials Requirements

Reactor	Power (MW(e))	Maximum Structure Temperature (°C)	Structure Material	Coolant	Breeder	Neutron Multiplier
BNL-I [IX-113]	1605	≈200	Al (SAP)	He	LiAl	Be
ORNL [IX-114]	518	1050	Nb1Zr	Li	Li	none
PPPL [IX-115]	2030	640	PE 16	He	FLIBE	(Be)
UWMAK-I [IX-111]	1473	500	SS 316	Li	Li	none
UWMAK-II [IX-112]	1716	650	SS 316	He	$LiAlO_2$	Be
UWMAK-III [IX-116]	1985	1000	TZM	Li	Li	none
LASL-ANL [IX-117]	4130	750	Nb1Zr	Li	Li	Be

Nuclear island

- structural materials;
- coolant;
- breeder;
- neutron multiplier;
- reflectors;
- shield;
- electrical insulator;
- magnet;
- thermal insulators;
- optical materials;

Balance of plant (BOP)

Structural:
- piping;
- tanks;
- heat exchangers;
- buildings;

Equipment:
- pumps;
- coolant;
- clean-up units;
- turbines;

Electrical:
- generators;
- transformers;
- wiring.

The categories under "balance of plant" are self-explanatory and will not be expanded here. Those under "nuclear island" cover a wide range of materials and requirements (see Table IX-XXII). The interesting feature about fusion reactors is that, at least with magnetically confined systems, materials must operate from liquid helium temperatures to temperatures as high as 1000°C. The needs range from pure metals to alloys, ceramics, covalent and molecular bonded materials and, finally, to non-metals.

The other feature about fusion reactors (at least magnetically confined systems) is their low power densities. This ranges from one to five MW/m^3 in TOKAMAK, Theta Pinch, or Mirror blankets, while for fission reactors the power density may be as high as 100 MW/m^3. This means that for a given size of power plant, fusion reactors will use *more* materials than fission plants. Since the rest of the balance of plant is roughly the same for fission and fusion, the extra cost of the fusion nuclear island must be offset by other parts of the system (i.e. fuel costs). Economic studies are just now realizing how critical this feature is to fusion power.

There are two other topics which should be examined in this section; these are:

- the use of solid breeders and the effect on types of materials required in the blanket;
- the use of enriched Li to increase breeding ratios.

The use of liquid Li as a breeding material as well as coolant results in rather large tritium inventories (of approximately ten kg) in the blankets which was thought undesirable. J.R. Powell [IX-113] proposed that one could use solid breeder materials (LiAl, $LiAlO_2$, Li_2O, etc.), which would allow the tritium to diffuse out as soon as it is bred. It was shown that by this technique tritium inventories in blankets could be reduced to less than 100 g. There was one drawback, however, the addition of other elements in the breeding zone reduced the number of neutrons available for breeding as well as the average neutron energy. As a net result, the Li6 fraction in the breeding zone has to be increased from its natural level of around 7.5 to roughly 90 per cent, in order to breed with the lower-energy neutron spectra. It was also found that, in order to compensate for the parasitic absorption and the reduction in the high-energy Li7 (n,Tn)-He reaction, one has to incorporate a neutron multiplier near the plasma. Such a requirement led to the inclusion of large amounts of Be or U in the blankets, both of which raise serious implications, one from reserves [IX-70] and the other from radiological considerations. Hence, in attempting to solve one problem, several more were generated. The more recent proposals to use Pb or Zr as neutron-multiplying media have yet to be closely examined in realistic systems.

3.3.2 Materials Inventory Per Reactor

The stated requirement for the seven reactors considered here are given in Table IX-XXIII for the "nuclear island", and in Table IX-XXIV for "BOP". All the results are quoted there as a function of elemental requirements, and *do not* (!) include replacement or burn-up amounts over an assumed plant life of 30 years. These will be treated later.

The results in Table IX-XXIII show a wide variance in requirements, which is mainly attributable to the detail with which the analyses were carried out. It is known that the UWMAK values include many components not considered by the other design groups, and therefore would be expected to yield the largest mass requirements. This is especially true in Table IX-XXIV, where the requirements for the nuclear island are normalized on the basis of ton per MW(e). The UWMAK numbers are consistently higher, owing to the more detailed nature of the study and the conservative nature of the power-cycle design. It is interesting to note that, except for the Theta Pinch reactor (where the true nuclear island is not well defined), the average material inventory is approximately 28 t/MW(e).

Table IX-XXIII: Summary of Original Materials Requirements--Nuclear Island[a] (t/MW(e))

Element	UWMAK I	UWMAK II	UWMAK III	PPPL	ORNL	BNL[b]	LASL-ANL
Al	0.24	1.33	2.42	0.01		1.70	
Be	-	0.25	-	0.43		0.12	0.01
B	1.07	2.50	0.47			3.61	
C	0.30	1.47	8.95	0.62	2.12	1.96	0.16
Cr	3.72	2.33	0.77	0.82	0.07	2.26	
Nb	0.07	0.09	0.05	0.05	1.93	0.07	0.87
Cu	5.06	6.11	0.65	1.61	0.87	4.59	2.25
He	0.04	0.04		0.06	0.002	0.04	0.30
Fe	12.79	7.96	2.56	5.87	12.4	7.76	
Pb	13.90	11.6	1.82		10.33	-	
Li[c]	1.15	0.35	0.20	0.29	0.89	0.05	0.38
Mn	0.41	0.25	0.09	0.08		0.25	
Hg	0.002	none	none			0.002	
Mo	0.39	0.24	2.82	0.14		0.23	
Ni	2.86	1.73	0.46	0.81	0.05	1.73	
Ti	0.04	0.05	0.04	0.01	1.24	0.004	
Sn	none	none	none	0.02			0.07
Zr	none	none	0.01				
F				2.62			
Total	42	36	21	13	30	24	4

[a] Assuming that all nuclear-island components last for the lifetime of the reactor, and no burn-up is accounted for;

[b] Using UWMAK-I magnet system;

[c] Natural equivalent.

The balance of plant numbers for the UWMAK systems are also given in Table IX-XXIV, as well as the summary of the toatl plant requirements for the UWMAK systems, on a per MW(e) basis. It is felt that these three reactors cover a sufficiently

Table IX-XXIV: Summary of Original Materials Requirements for UWMAK Reactors [a] (t/MW(e))

Element	Balance of Plant UWMAK I	Balance of Plant UWMAK II	Balance of Plant UWMAK III	Total UWMAK I	Total UWMAK II	Total UWMAK III
Al	0.30	0.01	-	0.54	1.34	2.42
Be	-	-	-	-	0.25	-
B	-	-	-	1.07	2.50	0.47
C	-	-	-	0.30	1.47	8.95
Cr	4.20	2.11	0.31	7.92	4.44	1.08
Nb	0.03	0.004	0.008	0.10	0.10	0.06
Cu	2.21	0.40	0.33	7.27	6.54	0.98
He	0.05	0.01	0.04	0.09	0.05	0.06
Fe	65.03	45.92	57.72	77.82	53.88	60.28
Pb	-	-	-	13.90	11.6	1.82
Li	-	-	0.04	1.15	0.35	0.24
Mn	0.41	0.21	0.37	0.82	0.46	0.46
Hg	-	-	-	0.002	-	-
Mo	0.01	0.05	1.26	0.40	0.29	4.08
Ni	3.05	1.45	0.21	5.91	3.18	0.67
Na	11.99	5.74	1.0	11.99	5.74	1.0
Ti	0.01	0.02	0.01	0.05	0.07	0.05
Sn	-	-	-	-	-	-
Y	0.003	0.002	$<10^{-5}$	0.003	0.002	$<10^{-5}$
Zr	0.07	0.06	0.002	0.07	0.06	0.01
Pd	-	-	<0.001	-	-	<0.001
Co	-	-	0.003	-	-	0.003
Total	87	56	61	129	92	83

[a] Assuming that all components last for the lifetime of the reactor.

wide range of TOKAMAK reactor designs, from lithium-cooled stainless steel to helium-cooled stainless steel with solid breeders as well as Li-cooled refractory metal systems, so that the average numbers are not so system-dependent. Here we find that BOP numbers average roughly 68 t/MW(e), and total plant numbers are an average of roughly 100 t/MW(e).

The envelope of maximum materials requirements for the construction of D-T reactors is given in Table IX-XXV for the

Table IX-XXV: Envelope of Maximum Material Requirements for D-T Fusion Reactors [a]

Element	Requirement (t/MW(e))	System
Al	2.42	UWMAK-III
Be	0.43[b]	PPPL
B	3.61[b]	BNL
C	8.95	UWMAK-III
Co	0.003	UWMAK-III
Cr	7.92	UWMAK-I
F	7.62[b]	PPPL
Nb	1.93[b]	ORNL
Cu	7.27	UWMAK-I
He	0.30[b]	LASL-ANL
Fe	77.82	UWMAK-I
Pb	13.90	UWMAK-I
Li	1.15	UWMAK-I
Mn	0.82	UWMAK-I
Hg	0.02[b]	PPPL
Mo	4.03	UWMAK-III
Ni	5.91	UWMAK-I
Na	11.99	UWMAK-I
Ti	1.24[b]	ORNL
Sn	0.07[b]	LASL-ANL
Y	0.003	UWMAK-I
Zr	0.07	UWMAK-I

[a] If all components lasted for the lifetime of the reactor;
[b] Nuclear island values only.

seven designs considered here. It should be noted that this column represents a worse case, and not all of these requirements would be present at the same time. On the other hand, those requirements for reactors other than the UWMAK series represent only the nuclear island values, and thereby may be lower limits.

3.3.3 Replacement, Enrichment, and Burn-up Requirements

It is generally accepted now that the first structural walls of CTR blankets will not last for all the lifetime of the plant. These walls will be damaged by the high-intensity-charged-particles neutron flux from the plasma, and will lose mechanical integrity in a matter of a few years. The following integral first-wall lifetimes have been assumed by the various design teams:

	$MW \cdot yr/m^2$
UWMAK-I	2.5
UWMAK-II	2.3
UWMAK-III	3.6
PPPL	8.8
ORNL	>10
LASL-ANL	10
BNL	3.8

Based on the various wall lifetimes and fuel-atom burn-up rates, we have calculated the annual replacement amounts on a ton per MW(e)·yr basis, and these are given in Table IX-XXVI. The low numbers in the ORNL case come from the *assumption* of long wall-life, and those for the Theta Pinch result from the *assumed* long wall-life and an extremely large power plant size.

It is interesting to note that both the BNL and UWMAK-II designs require the Be to be replaced and reprocessed in its radioactive state, and to be reinserted into the blanket. Hence, the replacement amounts here are mainly due to burn-up of the Be atoms and losses during processing. The BNL design also assumed that the solid breeder material will be reprocessed and refabricated to be reinserted into the blanket. The UWMAK-II study assumes that it will be too costly to reprocess such contaminated and radioactive material (impurities alone will make the ceramics extremely radioactive). It is also assumed that new, *enriched*, breeder material must be inserted into the reactor during each wall change. The numbers in the tables indicate the amount of *natural* Li which is required to supply this system, even so less enriched lithium will actually be in the reactor. Because of the high enrichment factor, this number actually exceeds the amount of lithium required in natural-liquid-lithium cooled systems over the anticipated 30 year plant

Table IX-XXVI: Annual Replacement Requirement of Materials Due to Radiation Damage in Fusion Reactors (t/MW(e)·yr)

Element	UWMAK I	UWMAK II	UWMAK III	PPPL	ORNL	BNL	LASL-ANL	Envelope
Al	0.005	0.210	-	-	-	0.157	-	0.210
Be	-	0.011[a]	-	-	-	0.007	0.002[a]	0.011
C	-	0.046	0.240	-	0.212	0.108	0.033[b]	0.240
Cr	0.088	0.088	-	0.019	-	-	-	0.088
Nb	-	-	-	-	0.193	-	0.019	0.193
Cu	0.005	0.005	-	-	-	-	0.270[b]	0.270
He[c]	-	-	-	-	-	-	0.004	0.004
Fe	0.301	0.304	-	0.042	-	-	-	0.304
Li	-	0.147[d]	-	-	-	0.01[d,e]	-	0.147
Mn	0.010	0.010	-	-	-	-	-	0.010
Mo	0.010	0.010	0.069	-	-	-	-	0.069
Ni	0.068	0.069	-	0.046	-	-	-	0.069
Ti	-	-	0.004	-	-	-	-	0.004
Total	0.49	0.74	0.31	0.11	0.41	0.27	0.33	

[a] For burn-up and 2 per cent reprocessing losses;
[b] If radioactive material cannot be refabricated economically;
[c] Leakage;
[d] Equivalent natural Li;
[e] Including 2 per cent fabrication losses.

life by a factor of almost two. Hence, it is obvious that the use of solid breeders may require *more* lithium to be mined from the ground than is required for the liquid Li case.

3.3.4 Projected Materials Needs

We have summarized the results of this analysis in Table IX-XXVII, and also listed the world resources of the various elements. One cannot simply assess the resources against the needs, until the amount of power generated by fusion is specified. Or, turning it around, one might specify the number of MW(e)·yr that could be generated by systems using those elements, if a few simplifying assumptions are made. These are listed below:

- 30 year plant life;
- All components of the nuclear island need to be replaced after 30 years, because they are either radioactive, or have worn out; material in the BOP can be used;
- All components that are replaced due to radiation damage cannot be recycled economically, with the exception of Be;
- The only makeup for Be is the burnup of the atoms themselves and for two per cent loss during fabrication;
- Requirements for Li are for the natural composition only;
- Helium can be reused;
- The total reserve picture remains the same as it was in 1976, that is, consumption of metals is just balanced by discoveries;
- Only reserves are considered at present prices.

The amount of MW(e)·yr that could be generated can be calculated by the following simple expression (except for Be and He)*:

$$\frac{\text{(Resource Limit at Present Prices)}}{\left(\begin{Bmatrix}\text{Initial Material}\\ \text{Charge in t/MW(e)}\end{Bmatrix}\cdot\frac{1}{24\ \text{yr}}\right)+\left(\begin{matrix}\text{Makeup Material}\\ \text{in t/MW(e)}\cdot\text{yr}\end{matrix}\right)}$$

* This analysis will use an 80 per cent plant factor, i.e. a 30 year plant life corresponds to 24 years of continuous electricity generation. We also assume that all nonradioactive components can be recycled.

Table IX-XXVII: Summary of Maximum Materials Requirements for D-T Fusion Reactors

Element	Maximum Requirement[a] (t/MW(e))	Maximum Makeup[a] (t/MW(e)·yr)	World Reserves[b] (10^6 t at present prices)	Maximum Energy Generated from Resource Limitations[g] (MW(e)·yr)
Al	2.42	0.157	2,000	$8 \cdot 10^9$
Be	0.43	0.011[c]	0.04	$1.4 \cdot 10^6$
B	3.61	0	20	$1.3 \cdot 10^8$
C	8.95	0.24	Large	Long
Cr	3.72	0.088	370	$1.5 \cdot 10^9$
Nb	1.93	0.193	7	$2.5 \cdot 10^7$
Cu	6.11	0.27	370	$7 \cdot 10^8$
He	0.30	0.004[d]	4.44[e]	$1 \cdot 10^9$[d]
Fe	12.79	0.304	276,000	$3 \cdot 10^{11}$
Pb	13.90	-	219	$4 \cdot 10^8$
Li				
(liquid)	1.15	$2 \cdot 10^{-3}$[f]	71.3	$1.4 \cdot 10^9$
(solid)	0.35	0.147		$1 \cdot 10^8$
Mn	0.41	0.01	72,000	$3 \cdot 10^{12}$
Hg	0.02	-	0.25	$3 \cdot 10^8$
Mo	2.82	0.069	13	$7 \cdot 10^7$
Ni	2.86	0.069	24	$1.3 \cdot 10^8$
Na	11.99	-	Large	Long
Ti	1.24	-	181	$4 \cdot 10^9$
Sn	0.07	-	3.7	$1.3 \cdot 10^9$
Zr	0.07	-	717	$2.4 \cdot 10^{11}$

[a] These are the maximum requirements as determined from the envelope of values for radioactive materials from Tables IX-XXIII and IX-XXVI;

[b] References [IX-111, and 112], world estimates as of 1976;

[c] Includes burn-up 2 per cent reprocessing loss;

[d] Leakage only;

[e] Reference [IX-118];

[f] Burn-up only.

[g] Without recycle of radioactive components.

These values are also given in Table IX-XXVII. We can class the results of this crude analysis with three categories as given below:

(1) Those elements which would allow one to generate $>10^6$ MW(e) for over 100 years--Al, B, C, Cr, Cu, He, Hg, Fe, Pb, Li*, Mn, Ni, Na, Sn, Ti, and Zr;

(2) Those elements which would limit the 100 year energy generation to present world electrical-generation level ($5 \cdot 10^5$ MW(e))--Nb, and Mo;

(3) Those elements which may be severely limiting to the long-term generation of fusion power--Be.

There are three main comments to be made here. First of all, the reserve picture for the 21st century is complete speculation; there may be more or less reserves depending on new discoveries and consumption rates. However, one might be forced to take the position that only roughly 10 per cent of the reserves should be allocated to any one field, in which case, B, Cu, Ni, Hg, and Pb fall into the second class, while Nb, and Mo fall into the critical case.

The second comment has to do with the allowance for price escalation, and thus the opening of much larger resources to the fusion field. This could certainly solve the problem for solid-Li breeding systems, because resources are three times the reserve levels. But it would not significantly change the problem for Be. The third comment has to do with local shortages of certain critical elements. For example, the US has essentially no Cr reserves so that, while the situation looks reasonable from a worldwide standpoint, the formation of cartels by countries controlling Cr would generate problems very similar to the present oil situation. Much the same could be said about every element, in that no single country abounds with them all so that it can satisfy its long term energy needs. Such a sobering outlook is not limited to fusion alone, and we expect that similar problems will exist for all the new and old energy sources.

3.4 Comparison Between Fission and Fusion Reactors

It is now of interest to place some perspective on the materials requirements for fusion and fission. We have already

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* Note that there is a factor of about three to four more electricity-generation potential with liquid-Li systems, even if one does not reuse the liquid in subsequent reactors. However, it should be relatively easy to clean up the lithium after appropriate decay of contaminants, and reuse it. This would extend the resource into the range of $4 \cdot 10^{10}$ MW(e)·yr.

treated the question of uranium reserves in Chapter II, and will not expand on that here. Furthermore, since the BOP for both the LMFBR and CTR should be comparable, we will focus our attention on the *non*-fuel materials requirements of the nuclear island.

We will assume a 30 year plant life for both systems, and replacement quantities as given in Tables IX-XVIII and IX-XXVI. Only major critical elements will be considered, and we will ignore the requirements for concrete, carbon, etc. The first comparison is given in Table IX-XXVIII, and reveals three major points:

Table IX-XXVIII: Comparison of Critical Materials Requirements for the Nuclear Islands of Fission and Fusion Reactors

	Initial Requirement (t/MW(e))[a]	Average Makeup (t/MW(e)· yr)[b]	Total Commitment Over Lifetime of Nuclear Island (t/MW(e))
Fission			
Steel	3.1	0.072	4.8
Sodium	2.81	-	2.8
Control Mat.	0.001	0.0004	0.01
Total	6.0	0.072	7.6
Fusion			
UWMAK-I	42	0.46	53
UWMAK-II	34	0.35	42
UWMAK-III	12	0.064	14
PPPL	13	0.092	15
ORNL	28	0.10	30
BNL	22	0.14	25
Average	25	0.20	30

[a] Initial amount only, only critical elements in nuclear island;

[b] Average over entire plant lifetime at 80 per cent plant factor.

(1) The initial materials requirements for the fission reactor is 6 t/MW(e) versus an average of 25 t/MW(e) for the six fusion-reactor systems considered. In the UWMAK series, each reactor was designed to require less material per MW(e) with UWMAK-III requiring only twice that of the LMFBR.

(2) The structural materials makeup for the LMFBR is more than a factor of three less than for the average of the six fusion reactors which is approximately 0.2 t/MW(e)·yr. However, recent reactor designs have addressed this problem, and in the UWMAK-III values were projected that are comparable to an LMFBR.

(3) The total amount of material (initial plus makeup) required over the 30 year life of the nuclear island ranges from 7.2 t/MW(e) for fission, to an average of 30 t/MW(e) for fusion reactors. Again, some recent fusion-reactor designs require only twice as much material as fission systems.

The obvious conclusion to draw from the above discussion--i.e., that fusion reactors may require two to seven times more material per MW(e) in the nuclear island than fission breeder reactors--is a direct result of the low-energy density in magnetically confined plasma devices. It is probably safe to say that there is no conceivable way that fusion reactors will require less material per MW(e) than fission reactors, even accounting for the possibility of higher thermal efficiencies.

The above numbers are tempered somewhat when the BOP values are added, which could range from 1.5 to 3 times the nuclear-island values. If we assume that, on the average, the materials in the nuclear island account for only one-third of the total materials requirement (see Table IX-XXIV), that fission and fusion BOP quantities are comparable, and that the building requirements are the same--the ratio of materials required for fission to those required for fusion ranges from 1.3 to 3 (an average of approximately 1.5). Such a ratio can have a significant effect on the cost of electricity through increased capital expenditures, as we shall see later.

Aside from the sheer mass requirements, it is also informative to examine the elemental requirements. This is especially vital for the nuclear island, where practically all the components will become activated or contaminated and may be removed from the reserve picture for thousands of years (see Chapter V).

The only non-fuel elements that appear to be critical in the LMFBR deployment are Cr, Ni, and Mo. Based on the reserve values in Table IX-XXI we find the following maximum energy limitations in the LMFBR scenario:

Cr: $8 \cdot 10^9$ MW(e)·yr

Mo: $3 \cdot 10^9$ MW(e)·yr

Ni: $7 \cdot 10^8$ MW(e)·yr

Of these elements, only Ni seems to be mildly constraining (remember that there will be other demands on the Ni resources, and it is unlikely that any energy source will be able to take more than roughly ten per cent of the total resources available), and even that could be offset by extracting Ni from ores which are uneconomical to mine at present. The concern about local shortages is still valid however, since the above elements are not scattered uniformly around the world. Nickel, molybdenum, or chromium cartels could cause a significant problem.

Finally, the increased demand for materials in fusion reactors coupled with neutron activation can have a significant effect on the cost of electricity from this energy source. For example, the present cost of high-quality, nuclear-grade, stainless-steel components installed in LMFBRs is roughly 20 to 30 $ per kg. At an initial inventory of 3.1 t/MW(e), this amounts to approximately 80 $ per kW(e) for the non-fuel part of the nuclear island alone. Assuming that this average $/kg number is also typical of all the mass in a stainless-steel fusion reactor*, one might expect approximately 300 to 600 $ per kW(e) for the same nuclear island in a fusion reactor. Such a challenge will be hard to overcome in view of world market prices for some elements, and future reactor designs should keep this in mind.

4. CONCLUSIONS

It is clear that one of the major factors limiting the efficiency and economic viability of both fission and fusion reactors is the degradation of materials performance in the reactor environment. The reduce component lifetime affects the economies of nuclear power plants in six major areas:

(a) reduced thermal efficiency (lower operating temperature);

(b) reduced plant factors (to change damaged components);

(c) increased capital costs (for remote handling equipment);

(d) increased operating costs (for component replacement and manpower);

(e) increased volume of radioactive wastes;

(f) increased demand for scarce elements.

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* Some components might be cheaper (i.e. Pb), but others would be much more expensive (i.e. Nb,Ti, Be,Mo, fabricated Li compounds, etc.).

These problems have been studied for over 20 years for the LMFBR and resulted in the choice of PuO_2/UO_2 as a fuel, SS 316 as a cladding and core structural material, and B_4C as a control rod material. However, it is quite probable that even these materials will not be sufficient for a completely economical breeder economy, and carbide fuels and high nickel-base alloys are being investigated for possible long-term application.

The process of selecting the optimum structural materials for fusion reactors is, by comparison, in its infancy. Early reactor designs almost exclusively used Nb alloy, but present thinking strongly suggests the use of austenitic steels at least for the first generation of power reactors. However, due to the high helium production rates, the pulsed nature of most viable reactor concepts, and the extreme reliability that will be demanded on the reactor components with respect to vacuum leaks and dimensional stability, it is now widely accepted that most of the reactor components will not last the lifetime of the power plant. The necessity to *quickly* replace damaged components in a very high radiation environment will put a severe strain on the design of a fusion power plant.

Both fission and fusion structural components and fuels share some of the same intrinsic radiation-damage problems. Void swelling in metals is probably more important in LMFBR because of the close tolerances for coolant flow, but high-temperature helium embrittlement will definitely be a greater problem for D-T fusion reactors than for fission because of the higher energy neutron spectrum. Irradiation creep will prove to be a major problem in both types of reactors because of: *(a)* high displacement rates in the LMFBR, and *(b)* the high thermal stresses in a fusion reactor. Fatigue is likely to be more severe in fusion reactors, especially in inertially confined systems.

Suggestions have been made to use alloys other than the austenitic steels in fusion reactors, mainly for the purpose of reducing the radiation levels in the wastes. However, essentially all of the proposed alloys suffer from one or more serious deficiencies in the fusion reactor environment. Aluminum alloys are subject to even more modest temperatures. The refractory metals Nb and V offer higher temperature operation with less irradiation induced embrittlement, but they are extremely susceptible to pick-up of interstitial impurity atoms, which also causes embrittlement, and they suffer from a lack of a commercial industry to supply a mature fusion economy. Molybdenum alloys are probably the best suited of the refractory metals but require major advances in joining of large-scale reactor components. For these and other reasons, it is of debatable validity to compare fusion-reactor systems based on these untested and ill-understood materials to workable LMFBR reactors. For the foreseeable future, fusion reactors based on austenitic steels will probably be the standard.

Because of the low power density in the blankets of all fusion reactors (not just the TOKAMAK), the nuclear-island requirement for materials is likely to range from 10 to 40 t/MW(e) of steel versus 3 t/MW(e) in fission breeder reactors. In addition, the replacement of damaged structural components may range from 0.1 to 0.5 t/MW(e)·yr of steel for fusion, compared to 0.06 t/MW(e)·yr for fission breeder reactors. These large material resource requirements for sometimes rather scarce materials will certainly be a greater problem in a fusion economy than in a fission economy. Careful attention will have to be paid to methods of reducing those requirements for fusion, or they could prove to be the limiting factor to the amount of energy that can be produced by fusion, despite essentially unlimited fuel resources. For example, one material which has been proposed for fusion reactors utilizing solid breeder blankets is beryllium. If this element is required, the ultimate amount of fusion generating capacity that can be built could be limited to as little as a few TW(e), and the electricity generated to a few hundred TW(e)·yr. (Total world generating capacity in 1976 is around 1.5 TW(e), and annual use around 0.8 TW(e)·yr.)

We conclude that materials problems are much more diverse and severe in fusion reactors than in fission reactors. Without intensive long-range development programs it is possible that fusion may never transcend the engineering feasibility phase into a commercial regime.

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X. WHAT WOULD BE REQUIRED FOR COMMERCIALIZATION? PROGRAMS, TIMING, AND FUNDING

W. Häfele, G. Kessler, G.L. Kulcinski

1. FAST BREEDERS--THE DEVELOPMENT OF THE ENTIRE FAST BREEDER SYSTEM

Timing and funding, as well as the construction and operation of test facilities for achieving scientific, engineering, and, eventually, commercial feasibility are the main elements of a development strategy for a new energy system such as the fast breeder. The target dates for entering commercial feasibility may differ for various countries, because they depend on the different energy resources available to them, and they must decide for themselves how to use these resources during the next decades. For example, there seems to be a clear-cut difference between Western European countries and Japan on the one hand, which have almost no oil or uranium ores of their own, and the USA and the USSR on the other, which do have their own coal, oil, and uranium resources.

Despite such different aspects, however, there seems to be the unanimous view of all countries in which fast-breeder reactors are developed that the liquid metal fast breeder reactor (LMFBR) system should enter its commercial phase between 1990 and 2000. The rest of the fuel cycle is expected to follow with a delay of 10 to 15 years. This has been reconfirmed in recent US [X-1] and European studies [X-2 to X-4].

With this target date of 1990 to 2000 in mind, we now discuss Figure X-1, which gives the timing and construction of both test facilities and power reactors from the early beginning in 1944. In doing so, we must cover the whole fuel cycle of fast-breeder systems and, consequently, consider the following fields of research and development:

- fast reactor physics;
- fast reactor safety;
- sodium-component technology;
- power-reactor development;

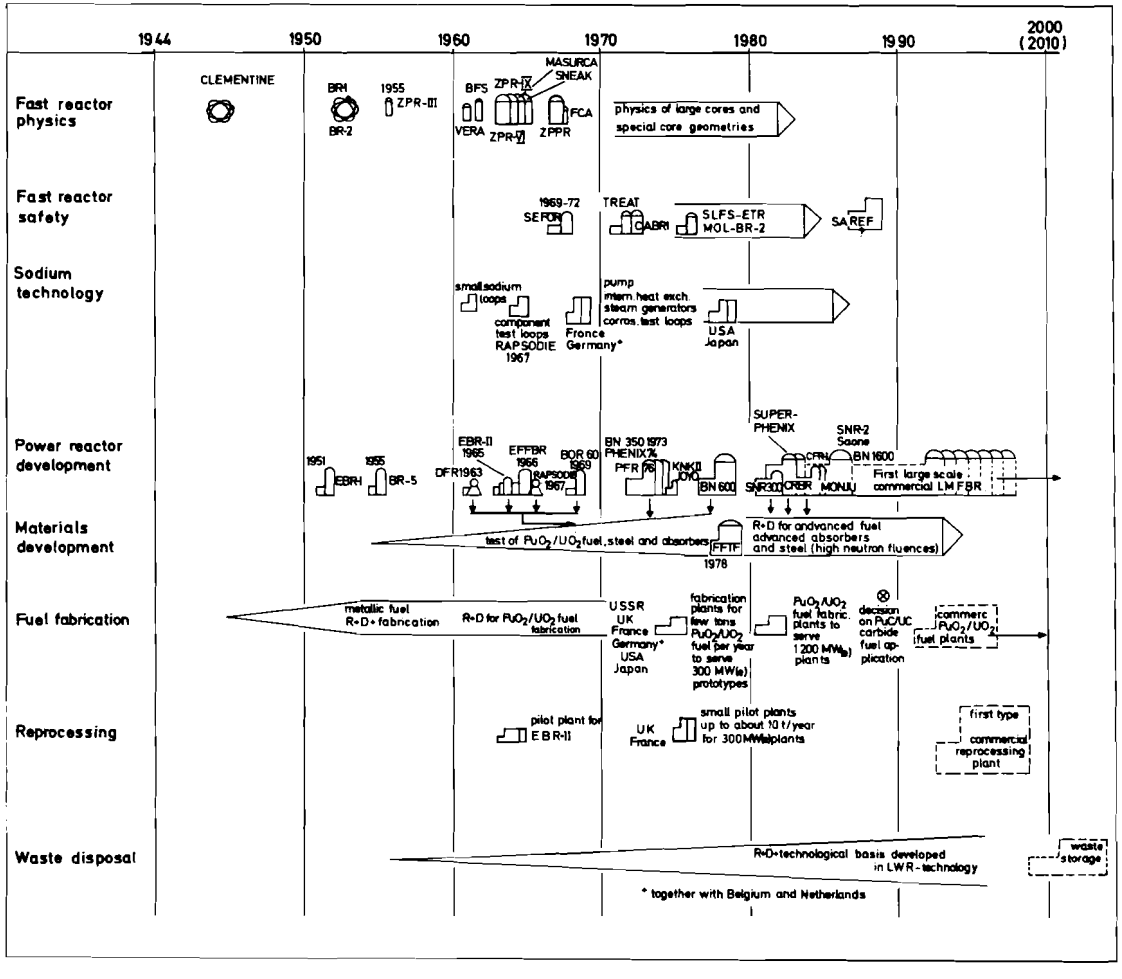

Figure X-1: Development Strategy for the Fast Breeder Reactor System

- materials development;
- fuel fabrication;
- reprocessing;
- waste disposal.

The principle of breeding was already recognized in the very beginning of nuclear reactor development. As early as in 1944 [X-5], E. Fermi and W.H. Zinn began to design a fast-breeder concept. Then (see Figure X-1), small physics and reactor test facilities were built (EBR-I, BR-5, and ZPR-III) to demonstrate the principles of fast-reactor physics and liquid-metal cooling. This demonstration phase for the scientific feasibility of fission breeding lasted for roughly 15 years, that is, until 1960. It was described in more detail in Chapter III.

Originally it was hoped to go on to power reactors by essentially one step. One must remember: In the mid-fifties, the normal commercial power station was not very large, it was only around 100 MW(e). It therefore seemed to make sense to head for the EBR-II and the Enrico Fermi fast-breeder reactor (EFFBR) in the US, and for the DFR in the UK. One must also recall that, under W. Cisler, foresighted utility groups were engaged in the construction and operation of the EFFBR. Their approach was also based on metallic fuel elements. The awareness of the limitations of that fuel concept coincided with the awareness of the rapidly growing unit sizes for commercial power stations. This led to the second generation of fast breeders. These fast reactors from the fifties and early sixties still exist and are now considered to be small. They were then used as test facilities and often converted to mixed oxides or carbide fuels. In this context, the early pioneering work of the Soviet BR-5 reactor must be mentioned. After early changes the French RAPSODIE reactor started right away with mixed oxides as fuel, and then followed the versatile test reactor BOR 60 in the USSR. The German/Belgian/Dutch program did not have such an experimental facility. But its participation in the multinational SEFOR project and operation of the liquid-metal-cooled thermal reactor KNK, which later was converted into the fast reactor KNK II, made up for that deficiency of the program. Significantly enough, it was felt necessary even with the Japanese program, which started much later, to build the Japanese JOYO reactor, an experimental reactor of that class.

For all fast-reactor groups of the world, the next target was then the design, construction, and operation of prototype reactors of the 300 MW(e) class. This required a broadening and sophistication of the various programs.

A number of zero-power physics test facilities were built (VERA, ZPR-VI, ZPR-IX, MASURCA, SNEAK, ZPPR, and FCA) for the further investigation of fast-neutron physics such as criticality effects and the sodium-void and Doppler phenomena, as

well as for the measurement of cross-sections and breeding ratios. Each country decided to build a zero-power test facility of its own. This was accompanied by the development of related large-scale software such as computer programs and cross-section sets, which were needed for the later design of 300 MW(e) plants.

A special case was the SEFOR project. The German/Belgian/Dutch fast-breeder project, in its association with EURATOM, joined forces with the US General Electric, 17 utility companies known as the South-West Atomic Energy Associates (SAEA), and the former USAEC. The objective of that program was to measure the Doppler coefficient as a principal safety feature of fast-breeder reactors of the second generation under operating conditions of a power reactor. But the objective meant more. It was geared not only to this measurement but also to the demonstration of the safety feature, and to the experience of related licensing procedures. The SEFOR program was successfully completed in 1971 (see Chapters III.1 and VII.1).

While the first power reactors (EBR-II, RAPSODIE, etc.) had been built without prior material test programs, it soon turned out that the fast-neutron irradiation and high neutron fluences of the envisaged second-generation prototype reactors required special material testing and development. This was somehow accomplished in the power reactors of the first generation, which then were converted into test facilities for Pu_2/UO_2 fuel, core-structural steel and absorber material. Know-how for PuO_2/UO_2 fuel fabrication was to some extent taken over from LWR technology. Reprocessing R&D work for PuO_2/UO_2 fuel was done in very small laboratory facilities. The last step towards demonstration of the technological feasibility, construction, and operation of the 300 MW(e) class of fast-power breeder reactors, could only be taken after sufficient experience with fuel-element irradiation. The RAPSODIE irradiation experience in France is one example. Before the core of PHENIX went into operation, about 20,000 fuel pins were irradiated to maximum burn-ups of about 15 per cent. There were similar requirements in other countries. While it was possible to investigate the physics problems with the zero-power test facilities of the mid-sixties, licensing discussions of the 300 MW(e) plants led to numerous out-of-pile test loops for the investigation of sodium boiling, sodium fuel interaction, aerosol behavior in containments, incore instrumentation, etc. In addition, in-pile programs in transient test reactors like TREAT and CABRI had to be initiated. The trend to request proofing experiments for licensing purposes has been expanding ever since.

For the design and construction of fast-breeder reactors of the 300 MW(e) class, it was necessary to engage industrial groups. Large sodium-component test facilities for pumps, heat exchangers, and steam generators were needed and tested prior to their operation within a 300 MW(e) plant. Their size is well within the range of earlier power stations, namely around 30 to 70 MW(th).

The funds required up to the end of this step, including the start of operation of a 300 MW(e) plant, are estimated to be roughly $ 1.8 • 10^9 for the German/Belgian/Dutch fast-breeder project [X-6]. The costs of the US LMFBR project activities until commissioning of the 350 MW(e) Clinch River breeder reactor (CRBR) are estimated to be up to $ 7 • 10^9 [X-7, and X-8]. Costs of the other LMFBR programs may range in between. The higher costs of the US LMFBR project are mainly due to its extended time schedule (early start, late target dates, e.g. for the operation of the CRBR) and its broad basis that includes numerous test facilities and programs.

While France and the UK are preparing for the last step up to a 1200 or 1500 MW(e) LMFBR plant, the USSR is continuing with another intermediate step, the BN 600. Construction of the SUPERPHENIX plant in France is scheduled to begin at the end of 1976. New component test facilities are planned for this step, since the extrapolation factor for its power output is about five as compared to PHENIX. The physics and safety questions are supposed to be covered by experimental programs under way in already existing facilities.

Within the next two decades the rest of the fuel cycle has to be developed further. Today all fast-reactor projects have small fuel-fabrication lines for PuO_2/UO_2 mixed-oxide fuel. In France, the present small fuel-fabrication line for PHENIX will be expanded to produce fuel for the 1200 MW(e) SUPERPHENIX plant. This fuel-fabrication plant will serve to produce up to 20 t of mixed-oxide fuel per year, starting in 1979/80. In the UK, the fuel-fabrication plant for the 250 MW(e) PFR has an annual production capacity of five to ten tons of mixed-oxide fuel. In the US, fabrication lines for fast-breeder fuel of about five tons per year are available. Similar capacities exist in the FRG, Belgium, and Japan.

These fabrication lines of 5 to 20 t per year will be capable of producing the fuel for the first 1200 MW(e) plant. As soon as more of these 1200 MW(e) units are being built between 1980 and 1990, the fuel fabrication lines will have to be expanded. As a rule, it can be assumed that a 1000 MW(e) plant needs 30 t of PuO_2/UO_2 core fuel every 2 to 2.5 years, which leads to a fast-fuel fabrication capacity of 12 to 15 t per GW(e)·yr. In addition, 15 to 20 t of UO_2 per GW(e)·yr will be needed for the axial and radial blankets. Decisions on carbide-fuel fabrication lines will not be taken before 1985 to 1995. If the decision is positive, fabrication lines with plant capacities similar to those for the oxide-breeder fuel will have to be built (the delay time of carbide fuel as compared to that of oxide-breeder fuel is about 15 to 20 years).

Breeder-fuel reprocessing plants with very small capabilities will go into operation in the UK and France within the next few years. For 1977/78, the UK plans to begin operation of a prototype reprocessing plant for fast-reactor fuel at the

site of the 250 MW(e) PFR reactor at Dounreay. The plant will have a capacity of about ten tons per year. Cooling times for breeder fuel will be 180 to 200 days after discharge from the reactor, before reprocessing can start. Storage capacities for cooling must then be made available. In France a similar (very small) reprocessing plant of about five to ten tons per year is in operation at Marcoule. In addition, the French intend to use their LWR reprocessing plant in La Hague with 800 t per year for some breeder-fuel reprocessing in an intermediate phase.

Both the UK and France plan the construction of a commercial-size reprocessing plant around 1990 that is large enough to handle the breeder fuel from 10 to 15 large-size fast-breeder plants. With the assumption that 20 t per year of core and blanket fuel are discharged per GW(e)·yr, these plants would have a capacity of 300 t per year. If the present fast-breeder development plan can be realized, such plants of 300 t per year would be needed between the years 1995 and 2000.

The disposal of highly active and medium-level waste will not become a problem for fast breeders before 1990 until the installation of fast-breeder plants total a power of some 10 GW(e). The LMFBR system can then take advantage of the methods and techniques developed for LWR fuel wastes.

At present, discussions, mainly in the US, center around the speculation that the licensing of the first-of-its-kind 1200 MW(e) LMFBR might require a number of large-scale safety tests. Therefore, plans for the construction of a safety reactor facility (SAREF) are being discussed. In the US, materials research is also following on a broad scale in the fast flux test facility reactor (FFTF), whereas the European countries rely on irradiation tests in their fast breeders of the 300 MW(e) class.

It is difficult to define the beginning of the commercial phase of LMFBRs. The first 1200 MW(e) to 1500 MW(e) plants will still require additional R&D and governmental support since they cannot be built and operated on a fully commercial basis, competing with commercial LWR plants of the same size. Only with some delay (10 to 20 years) will it be possible to develop the rest of the fuel cycle to the same or an equivalent size. This is so because sufficient quantities of irradiated fuel must pile up before large-scale reprocessing plants with through-puts of 300 t per years, and later on 1500 t per year, can operate. It is estimated that about $\$ 10 \cdot 10^9$ (in 1975 dollars) will be needed in the US up to operation of the first plant of 1200 MW(e) to 1500 MW(e) (Figure X-2). This amount implies a broad R&D program like the one that is being performed and planned in the US. This would include the FFTF, CRBR, and part of the first 900 MW(e) to 1500 MW(e) plant to be built after 1990 [X-8, and X-9]. But it would not include the development phase until the fuel cycle is closed and operated on a commercial basis.

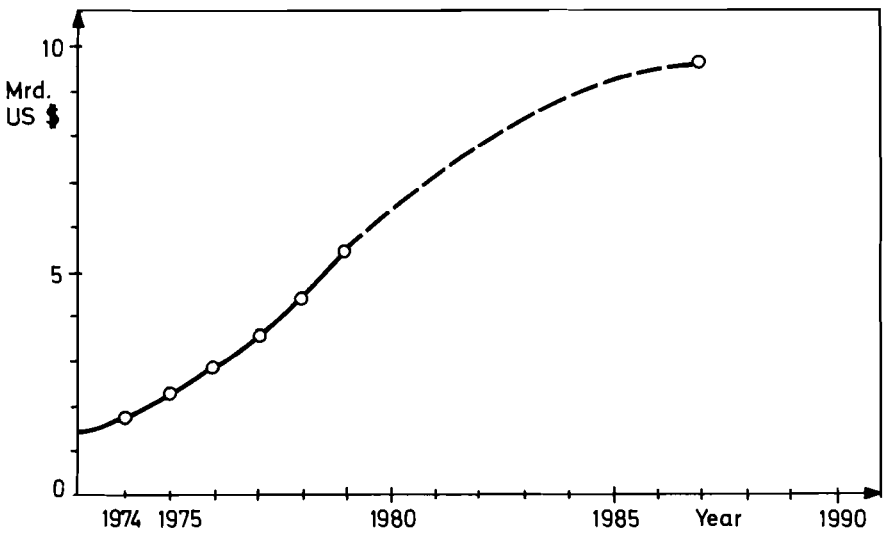

Figure X-2: Cumulative Total Cost Projection of the US LMFBR Program

2. SOME PROBLEMS OF NUCLEAR ENERGY PENETRATION

(A.M. Belostotsky)

As was mentioned earlier, except for the coal option, breeder and fusion technologies are the main means of transition from limited to unlimited energy for mankind. In discussing these two technologies, we are referring mainly to the production of electricity. Such an orientation towards new technologies, however, would not eliminate a future shortage of conventional fuels in the world, since electricity does not cover the full demand for energy.

To illustrate the importance of electricity in the total present energy consumption, Tables X-I and X-II are given which refer to the Soviet Union and the United States. These tables show rather low shares of electricity in the total energy consumption in both cases: 18 and 25 per cent of primary energy are used for electricity generation, respectively.

Table X-I: Utilization of Energy Resources in the USSR, 1970 [X-10]
(%)

(1) Power Plants	36
Electricity generation	18
Heat generation	18
(2) Devices for Direct Fuel Utilization	52
Engines and machinery	19
Industrial furnaces	20
Space-heating furnaces and appliances	13
(3) Boilers	12
Total	100

Table X-II: US Consumption of Energy Resources by Major Consumption Sectors, 1972 [X-11]
(%)

(1) Electricity Generation	25
(2) Household & Commercial	21
(3) Industrial	29
(4) Transport	25
Total	100

To make a real transition from fossil to nuclear energy, it will be necessary to widen the sphere in which nuclear energy is implemented, broadening it to include new branches of the economy.

Several steps have been taken in the Soviet Union to include in plans for future energy development the use of certain amounts of secondary energy in the form of heat for industrial and domestic uses (heating and warm water supply) [X-12]. As can be seen from Table X-I, the use of nuclear energy to produce heat for secondary purposes permits to increase the total nuclear energy production by roughly a factor of two.

The economic reasons for this are not proportional, however. But we shall put aside for now a number of details regarding the commercial feasibility of producing heat from nuclear energy, and producing electricity and heat at the same power plant (this strongly increases the economic efficiency of such a plant--some assessments of this problem are in [X-12]).

It should be noted that the total amount of heat available on the basis of nuclear fuel is less than mentioned, because the supply of small and separate consumers is probably not quite efficient. Nevertheless, the role of nuclear energy in the overall energy production will be much greater if it includes heat production.

In the case of the Soviet Union, there are some additional advantages: Nuclear energy for heat production conserves oil fuels, and, moreover, the use of nuclear energy for electricity production saves coal. As for the United States, there are probably not so many private consumers concentrated in one place, and the heating supply from centralized power plants would often not be efficient from an economic point of view.

Another example of a multi-purpose installation is the BN 350 breeder-reactor power plant (see Chapter IV.1.2).

The role of nuclear power plants, and especially of those with fast-breeder reactors, is not limited merely to the production of electrical energy and heat. They are also of interest for water desalination combined with heat and electricity production.

About 60 per cent of the earth's surface is dry, arid, or semiarid. In order to make these areas agriculturally productive and satisfy the demands of an increasing population, great quantities of freshwater will be needed. Yet in the meantime the world's supply of freshwater is decreasing as a result of high levels of consumption. A portion of man's water resources is irretrievably lost because of his agricultural activities.

Water desalination is developing at an ever increasing rate, with an expanding number of desalination plants: In 1962, the world-wide capacity was 75,000 m^3/day, and in 1966, 370,000 m^3/day; toward the end of 1975, the hypothetical total capacity of all the world's desalination plants reached 3.5 million m^3/day.

In the Soviet Union, much experimental work is in progress, and many water-desalination methods are being studied. The most efficient and widely used desalination technique in the USSR is the distillation method.

It is precisely this method that is at the heart of the large industrial desalination installation in Shevchenko in the Trans-Caspian region. This plant may serve as an example of a certain unexpected economic profitability of combining the electricity production at nuclear power stations with water desalination. It turns out that nuclear plants are very economical when operating at a high and fixed capacity and simultaneously producing electricity and large quantities of desalinated water. This is being done in Shevchenko. The total output of all the installations, using distillation by evaporation, of this plant is 100,000 m^3/day. If one takes the average daily expenditure of water per person to be 600 liters then the output of this complex is sufficient to supply a city with 160,000 inhabitants (not taking into account the demands of industrial activities).

A dual-purpose nuclear plant can operate with a high load coefficient. Here one may be very flexible in combining electrical and desalination operations. At peak periods, the full output of reactor-generated steam can be directed to generate electricity, whereas for periods when the demand drops, the nuclear plant may direct unneeded heat to the desalination complex and collect the resulting freshwater in closed or open reservoirs.

In view of the need for freshwater and its growth, the economic advantages of dual-purpose nuclear plants for electricity production and water desalination are clearly evident.

All this makes the problem of increasing the use of nuclear plants more complex and specific for different regions and countries. On the other hand, much can be done in the sphere of the national economies to prepare them for the energy supply peculiarities of nuclear energy. For example, most energy consumption in the Soviet Union occurs in its European regions, whereas the country's main energy resources are situated in Siberia. Because of transportation, the energy cost in the European regions is much higher than in Siberia. So the leading economic policy for a long time has been to build industrial facilities in Siberia that are rather energy intensive. Nuclear energy will decrease the energy cost in the European regions as compared to that in Siberia, although it cannot make them equal. Moreover, it will change the orientation of industrial building in Siberia. Thus reasons other than energy cost will become most important in the choice of how industries are to be distributed throughout the country.

Changes in the relative costs of different energy resources will also influence the efficiency of their use in various branches of the economy.

So the global direction of energy growth involves a transition from conventional resources to nuclear energy on the basis of fission breeders, with fusion reactors representing a further stage and thermal reactors the first stage of nuclear energy development.

When investigating the development of nuclear energy it is important to study the penetration of second-stage nuclear technologies into energy systems. In this connection it is useful to mention the following aspects of the problem.

There are different points of view on Pu breeding in fast reactors. Most USSR scientists believe that Pu breeding in operational fast reactors must satisfy the needs of:

(a) the first loads of new fast reactors under construction;

(b) reloading of fast breeders in operation;

(c) partial or even full substitution of U235 in thermal reactors.

A. Alexandrov has assessed (see Chapter II) that under such conditions breeders will have to double their four to six year period (i.e. eight year period), in order to double the total electrical capacity*. For a good start, fast-breeder reactors will have to be supplied with the definite quantity of Pu that has been stored during the operation of a system of thermal reactors. Also, it is possible to store Pu in some special ways, for example with the help of thermal converters.

For maximum Pu storage it is desirable to use all reactors with maximum power plant factors. But peculiarities of electrical energy demand make this impossible. Maximum efficiency in distributing reactors according to the power plant factor is another economic problem.

All this necessetitates a definite strategy for reactors, nuclear fuel supply and conservation, and reactor use in the power system. And all these problems very much influence the commercial feasibility of nuclear energy; they must be carefully investigated with the help of mathematical models. Some steps in this direction have been taken already [X-13 to X-16], but greater efforts must be made to solve these problems, especially with respect to the peculiarities of the development of nuclear energy in different countries (or regions).

Nuclear fuel supply and the nuclear fuel cycle add another dimension to the nuclear energy economy.

The capital investments needed for building a nuclear fuel plant are very high and, therefore, not available to every country in the world. Thus the integration of different countries in developing a common nuclear-fuel industry seems

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* This strategy is distinctly different from the strategy in Europe and the US that has evolved most recently. By contrast it concentrates on the use of the Pu that is converted in present LWRs. The breeder acts largely as a disposal system and, instead of the doubling times, it is the first core inventory which becomes the leading parameter.

inevitable. The economic grounds of such an integration should be investigated. Due to the high costs of nuclear fuel facilities, the development of the nuclear fuel industry depends very much on the timing of the transition from thermal reactors to breeders. This is so because of the differences in fuel fabrication technologies for thermal reactors and breeders.

As was mentioned above, breeders do not need certain technological processes typical of thermal reactors such as uranium enrichment, and they need some of the other processes in smaller quantities (mining of ore and milling). So the problem is: How to develop the nuclear-fuel industry with maximum efficiency during the transition period while waiting for breeders (or even fusion reactors)?

Difficulties in solving all these problems are rapidly increasing, especially in view of the fact that decisions must be taken under great uncertainty as to timing, economy, resources, labor and material needs, and so on.

3. FUSION REACTORS--THE FORMULATION OF A MASSIVE PROGRAM BEFORE THE EXACT CONFINEMENT CONFIGURATION HAS BEEN DETERMINED

The present state of the art of plasma physics and fusion technology does not allow one to be too detailed about the performance of all the necessary facilities that will be needed for the commercialization of fusion. However, we can reflect on the progress made since the 1950s, and suggest the minimum framework of test facilities that would be required for commercialization decisions. Such a plan for the TOKAMAK is illustrated schematically in Figure X-3, and one could conceive of a similar plan for Mirrors or inertially confined systems.

It should be recognized that, although the scenario in Figure X-3 is developed only for the TOKAMAK, there will be, and has been, considerable contributions from the other lines of plasma research. Since the first proposal of a stellarator in 1951 [X-17], a vast amount of research has been performed in the area of confining and understanding hydrogen plasmas. After a great deal of frustration in the 1950s and early 1960s, a major breakthrough was made by the USSR in 1963 in the form of a new confinement device called a TOKAMAK. After some original experiments with a device called TM-3, a large device called T-3 was built, and in 1968/69 the results from T-3 convinced the world that there was indeed some promise in such a confinement scheme. This device was later modified and is now called T-4. Very soon after the T-3 results, physicists at the Princeton Plasma Physics Laboratory (PPPL) converted the Model C stellarator to the ST-TOKAMAK, and in 1970 confirmed the positive results of the Soviets.

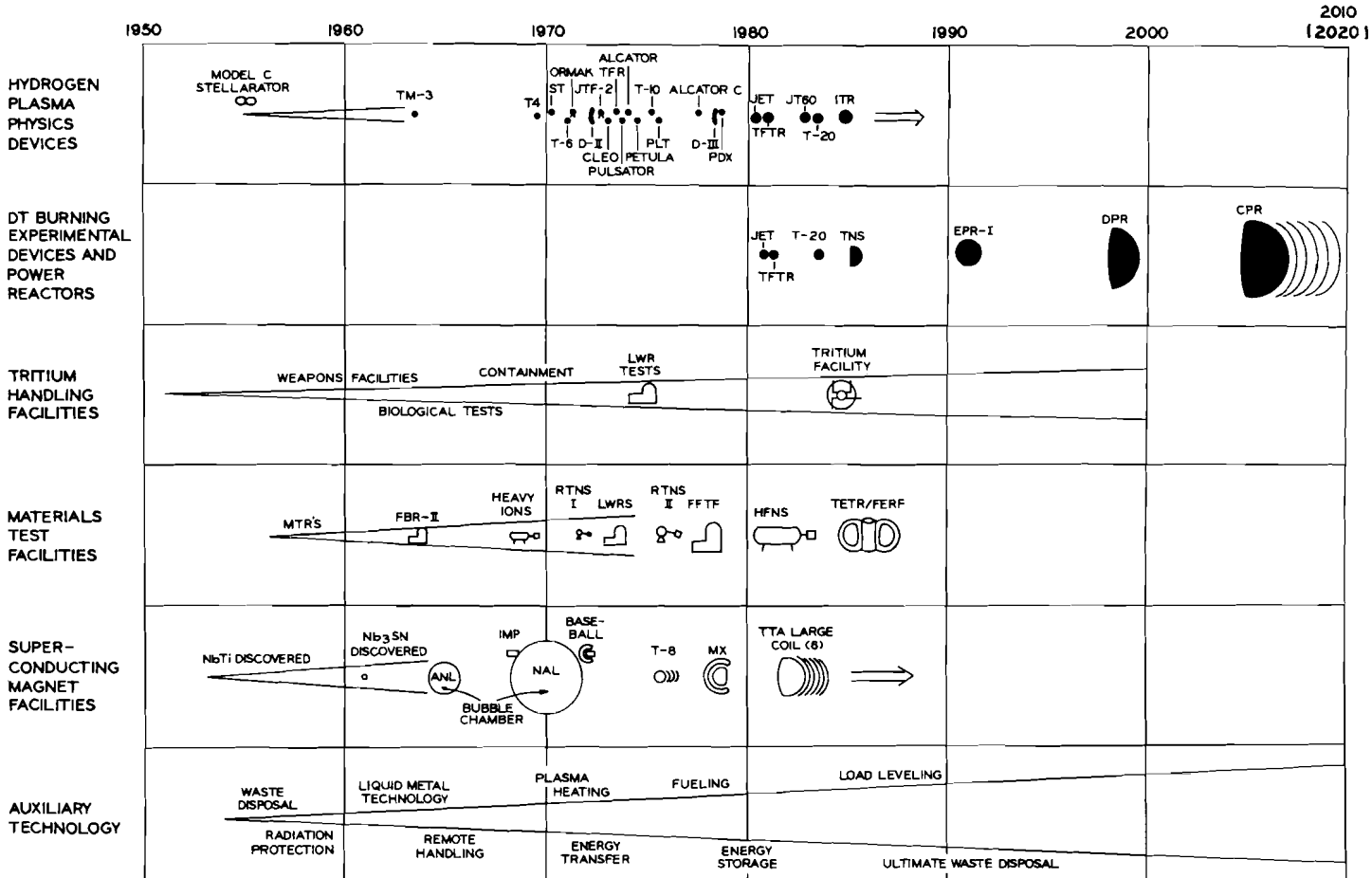

Figure X-3: Development Strategy for a TOKAMAK Commercial Reactor System

The period of 1970 to 1975 saw a great deal of activity in the TOKAMAK area with the construction of larger circular plasma devices in the US (ORMAK, 1971; adiabatic toroidal compressor (ATC), 1972; ALCATOR, 1973, and the Princeton large torus (PLT), 1975), the Soviet Union (T-6, 1971; TO-2, 1972; and T-10, 1975), the United Kingdom (Cleo, 1972; Petula, 1974; and DITE, 1974), France (TFR, 1973), Italy (Frascati, 1975), the FRG (Pulsator, 1973), and Japan (JFT-2, 1972). The effect of non-circularity was originally tested in DOUBLET-II at General Atomics in the US about 1972. All of these devices are aimed at testing the scaling laws, i.e. the effect of plasma size (current) on confinement time, the limits of ohmic heating in raising the plasma temperature, the effect of impurities, etc. Auxiliary technologies have also been tested on these devices, such as neutral beam or RF heating of plasmas, fueling, feedback control, impurity control, and a limited amount of superconducting magnet design.

Future major US TOKAMAK hydrogen burning devices include the DOUBLET-III (1977) to test scaling laws in non-circular shapes, the poloidal divertor experiment (PDX) to test methods of removing impurities from plasmas and to protect the first wall (1977), and the ISX (1978) to test various non-dynamic means to reduce contamination. The JT-60 in Japan (1982) will also contribute to the understanding of plasma scaling laws in hydrogen plasmas.

The first major D-T burning facilities are scheduled for the 1981 to 1984 period. The US TOKAMAK fusion test reactor (TFTR) is scheduled to operate in 1981 and will be, in addition to having the largest plasma radius and volume up to that time, a major test on the ability of a plasma to have up to 30 MW of power injected into it without introducing any major instabilities. This will not be an ignition device, but will operate in the "driven" mode. The first ignition device to burn tritium will be the joint European torus (JET) scheduled for operation in 1981. Not only will this device test plasmas with dimensions within a factor of two or three of commercial devices, but it will also have the ability to do so in non-circular plasmas. Finally, an even bigger plasma volume (approximately 400 m^3) will be tested in the Russian T-20 in 1984/85 with roughly 50,000 D-T pulses per year. Reliability at high repetition rates will be one of the main objectives of this device, in addition to a major test of plasma scaling to near-commercial reactor size. All of the above devices will use normal (copper) magnets.

With some experience on the α particle effects in D-T plasmas, there will undoubtedly be one or more devices required to make the final step up to reactor-grade (size) plasma with superconducting magnets, before power reactors are actually built. Such devices will occupy the 1985 to 1990 time period, and be followed by the first experimental power reactors (EPR) of the early 1990s. The EPRs would not only be confirmations of the scaling of plasma-physics laws, but they would experience

significant radiation damage, heat generation, probably have to produce a substantial fraction of the tritium they burn, and be coupled to an electrical power generating facility. Such reactors, while producing large amounts (several hundred thermal megawatts), probably would not do so on a net energy balance basis, and would not be required to demonstrate high plant factors.

Once the major technology problems have been tested and demonstrated on the EPRs, the next series of reactors in the late 1990s would attempt to combine all of the features of a power reactor into one operating facility. These demonstration power reactors (DPR) will have to produce net power in the several hundred MW(e) range, produce more tritium than they consume, demonstrate high reliability and plant factors greater than 60 to 70 per cent and demonstrate that emission of radioactive effluents can be controlled and final disposal of radioactive blanket components can be safely achieved. This DPR does *not* have to be economic in a competitive sense, but it must demonstrate the ability to be economic in subsequent commercial power reactor (CPR) designs. The CPRs will probably operate no sooner than five to ten years after the first successful DPR (roughly 2005 to 2010).

Turning to the auxiliary technologies that have to be developed, we first observe that these facilities, in total, may even exceed the cost of the first EPRs. Let us consider a few unique technologies that must be developed further than their present status.

Obviously, a great deal of work must be done to establish safe and effective handling of the tens of kg of tritium that may be present in a fusion reactor. There is a large body of knowledge that already exists from the various weapons programs around the world on breeding, extraction, and handling of tritium. This work started in the 1950s, and there are over 20 years of experience to build on. In the mid to late 1970s, several small scale studies in LWRs have been and will be initiated to test the specific behaviour of CTR breeding materials, and plans are now being laid for the 1984 operation of a large-scale tritium handling facility. The experience form TFTR, JET, and T-20 should combine with that from the LWR tests and the tritium facility to provide a sufficient base for T_2 handling in EPRs.

The material research required for the fusion program is probably the second most difficult problem (next to plasma physics) to solve before commercialization can be realized [X-18, and X-19]. A certain amount of ground work for high temperature and high damage effects in steels has already been established from the fast-breeder program and selected irradiations in light-water reactors. On the other hand, very little has been done on the high-fluence testing of refractory metals in these facilities. A crude, but reasonably satisfactory

theory now exists to explain high-temperature neutron radiation damage in metals, but very little information is available on the effects of helium (see Chapter IX), or on the dimensional properties of non-metals like graphite. The effects of damage rate on one property, void swelling, is currently being studied by high-energy heavy-ion bombardment, and this activity will probably continue for many years to come.

The first data on 14 MeV neutron low fluence (lower than 10^{17} n/cm^2) have evolved from the rotating target neutron source (RTNS) at the Lawrence Livermore Laboratory, and a higher-flux device is now being constructed. This latter device will be dedicated to fusion materials, and should allow a small number of materials to be studied at fluxes of roughly 10^{13} 14 MeV neutrons/cm^2•sec, but only on a small volume (roughly one to two cm^3). Since many data points will be required for qualifying fusion materials, a much larger volume (and flux) would be desirable. Such a device is currently being planned in the US for operation in 1982; it is called the high flux neutron source (HFNS), and utilizes the D-Li reaction. A volume of approximately 100 cm^3 will provide a flux greater than 10^{14} n/cm^2•sec of neutrons with E > 10 MeV. This is equivalent to roughly 2.5 MW/m^2 of uncollided flux. However, even the HFNS will not be sufficient to supply all the information needed to build a DPR, and a much larger volume facility (approximately 10^6 cm^3 of high-flux zone) will be required. Proposals for such facilities have been made with the TOKAMAK engineering test reactor (TETR) [X-20] and FERF [X-21] in the US, satisfying most of the goals for this program and also providing valuable engineering experience for the TOKAMAKs of the 1990s. Finally, there will probably be a need for an advanced materials test facility in the 1990s to optimize alloys and other solid materials (breeders, n-multipliers) for commercial reactors.

The development of S/C magnets for the fusion program has a very small base from which to start. High flux coils (>40 kg) have only been constructed in one meter bore up to 1976. Since 10 m bore coils will be needed for EPR, and roughly 15 to 20 m for DPRs and commercial systems, a large and, up to now non-existent, program needs to be put into place. The first step towards such a development is the establishment of the technology test assembly (TTA) to operate in 1982. The goal of such a facility is to fabricate approximately six superconducting coils of 5 m bore, which can be used to extrapolate to the EPR. There will also be a need to encourage industry to establish the capability of producing approximately 10 to 20 m bore magnets in the 1990 to 2010 period, and eventually scale up to approximately several hundred such coils per year after 2010.

Finally, there is a host of smaller but vital technologies that must be developed. Plasma heating methods (i.e. neutral beams, RF heating) are currently being adequately supported for the near-term devices. On the other hand, fueling, which will be absolutely essential for the power reactors, has not been adequately supported, and much more work needs to be done

in that area over the next ten years. Energy transfer and storage are currently being studied, but a much larger effort is needed to establish their commercial feasibility. Eventually, methods of load leveling of power output will have to be formulated and tested before the DPR can operate. Waste disposal techniques, radiation protection, and liquid metals technology for fusion will be largely developed by the fission industry. This work has been ongoing for roughly 10 to 20 years, and the liquid metal technology should be a simple extension of the recent LMFBR work. However, a decision on the reprocessing and reuse of radioactive components will have to be made towards the end of the century so that economic assessments of materials resources (see Chapter IX) can be made.

All of these programs will cost a great deal of money, and we have plotted in Figure X-4 the cumulative expenditures for fusion in the US versus time as projected by ERDA [X-22]. Note that from the present cumulative values of 1.7 billion to the 15 to 20 billion (in *logic III*) in the year 2000 requires a vigorous effort, far in excess of its present level of support. The estimate for the fission breeder is also shown in Figure X-4; it reveals that the fusion costs will be roughly comparable to or slightly higher than the commercialization cost of fission when the associated fuel reprocessing and storage facilities are built. A final word of caution to the reader is that Figures X-3 and X-4 should not be construed as saying that fusion reactors could (or should) replace the LMFBR simply because the schedules are close and the funding comparable. The element of risk is much higher in fusion because of the unknowns in plasma physics and materials performance. A prudent approach would be to develop both technologies in parallel, at least up to the year 2000 before making any long-term decisions.

4. CONCLUSIONS

It is expected that it will take 50 to 60 years to pass through the scientific and engineering feasibility stages before demonstrating commercial feasibility of the fast breeder reactor starting from 1942. This is based on the belief that commercial feasibility of fast breeder power stations could be attained between 1990 and the year 2000, while the related fuel cycle services are expected to require an additional 10 to 15 years.

Approximately the same time frame is anticipated for fusion reactors (although the uncertainty is greater): commercial feasibility of large-scale power stations could be achieved in the time period of 2010 to 2020. Unlike the LMFBR, there are no fuel-cycle-related services which will be required after a large number of fusion reactors are constructed, and, therefore, the point of commercial feasibility should be easier to define.

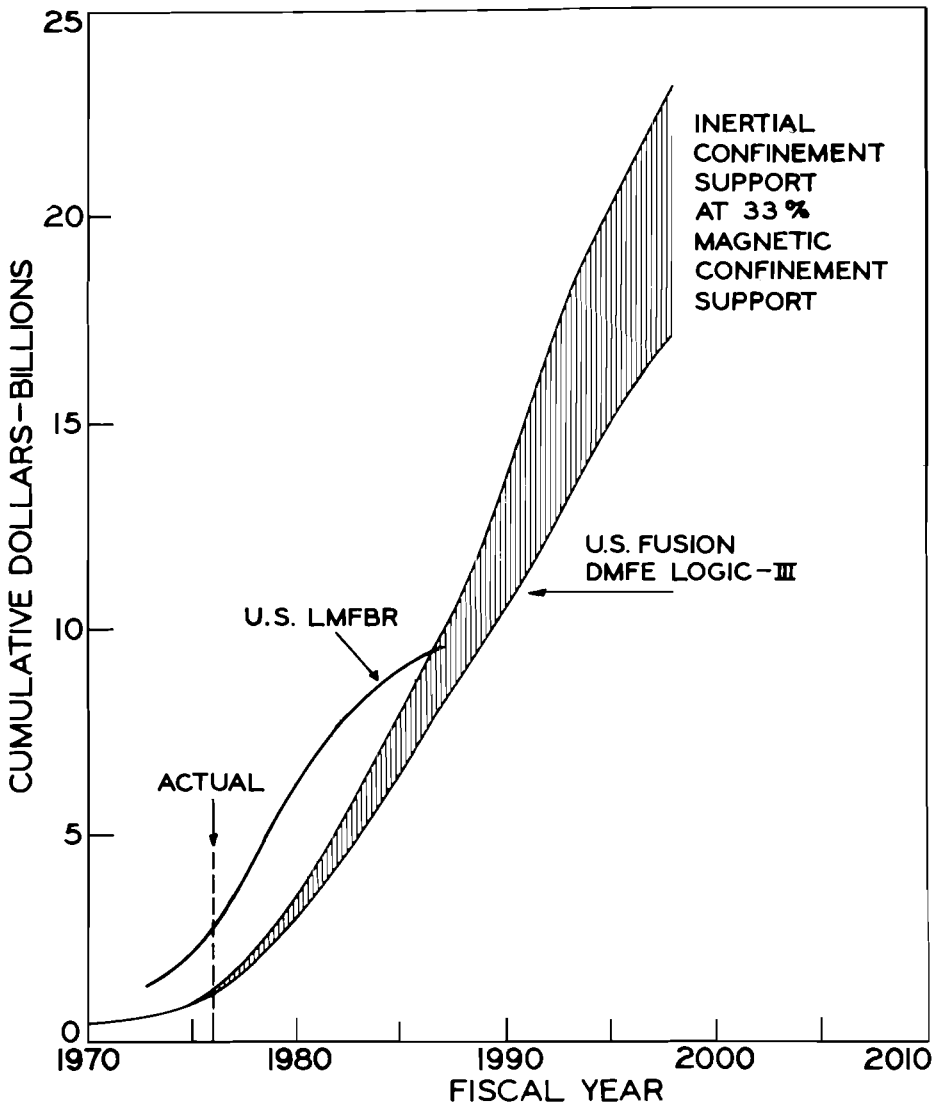

Figure X-4: Projected Cumulative Cost of LMFBR and CTR Commercialization

Three generations of reactors seem to be required for both types of energy sources to demonstrate commerical feasibility:

- experimental power reactors (10 to a few 100 MW(th));
- prototype or demonstration reactors (250 to 500 MW(e));
- semi-commercial reactors (1000 to 1500 MW(e)).

Along with these major facilities, there are a large number of smaller but equally important test facilities that need to be developed for:

- physics;
- engineering;
- materials;
- safety.

Beyond the physics facilities, the materials testing facilities can be particularly costly and time consuming to the overall program development. There is hardly any way to circumvent these problems, as each generation of reactors requires higher performance characteristics which are difficult to test in facilities existing up to that point. In order to be useful, fast breeders and fusion reactors must fit into existing schemes and rules of electricity production. Demonstration of availability, maintainability, and repairability is in itself a complex procedure that requires time. Especially important is the aspect of licensing. In fact, the rules and fundamental data underlying the licensing process must be developed almost in parallel with the reactors and facilities that are so to be licensed. Aspects of public acceptance broaden the scope even further.

Finally, the fuel cycle that serves fast breeder reactors must also be developed. By necessity, the hot part of the fuel cycle can be developed on a technically significant scale only when irradiated fuel is available in significant quantities from reactors whose acceptability often seems to require the services of such fuel cycle. In fact, the LWR seems to be presently in such a situation. The problems of final waste disposal extend the time horizon even further.

In the US, with a broad and stretched-out development program, more than ten billion dollars are expected to be necessary for reaching commercial maturity of LMFBR. By contrast, in European countries, the development programs seem to be less broad and less stretched out and, thereby, seem likely to be considerably cheaper. The difference in funding points to the degree of flexibility that such programs seem to have. In any event, it must be borne in mind that there are parallel develop-

ment programs in the world whose positive interactions contribute significantly to each other. The value of such positive interactions is rather high. In fact, that may explain the lower cost that European programs seem to have when compared with the US program.

The situation for the fusion program is much less well defined, but recent projections in the US program reveal that it may require 20 to 25 billion dollars to bring fusion through the demonstration power reactor phase, and it is not unreasonable to expect that another five to ten billion dollars will be required to progress through the commercialization stage. In contrast to the breeder program, the European fusion program is much smaller and of a longer time duration. The Soviet program is approximately the same as the US program in level of effort now, and it is expected to keep pace with the US program. Therefore, it is reasonable to expect that--worldwide-- it may require as much as 50 billion dollars to reach commercial feasibility of fusion. The same benefits for international cooperation in fusion research as in fission are expected to allow for considerable flexibility in design and should increase the probability of long-term success.

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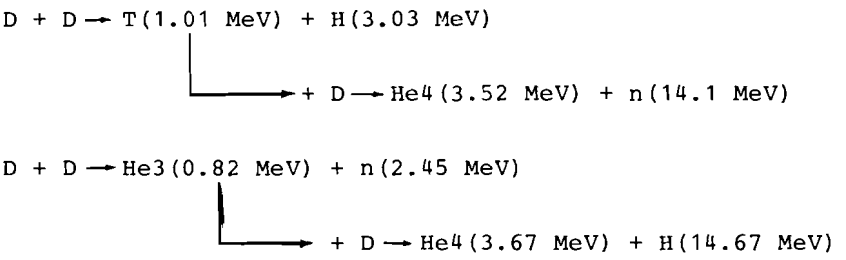
APPENDIX A

THE POTENTIAL ROLE OF D-D REACTIONS IN FUTURE FUSION DEVICES

G.L. Kulcinski

In spite of the fact that the D-T reaction is the "easiest" of all the thermonuclear reactions to initiate and contain, it does have some features that have prompted scientists to look for an advanced fuel cycle. The two major problems are the control of large amounts of tritium and the severe radiation-damage effects. The most likely improvement of the D-T cycle will be the D-D cycle described below with the energies given in parentheses.



The major advantages of this system are that the fuel resources (deuterium) are truly unlimited (see Chapter II), and that there is no requirement to breed tritium in the blanket. The resource question is of limited importance as the Li reserves for the D-T cycle are also quite large, but the effect on reducing the "true" fuel cycle costs for fusion should be very large. This latter statement would be especially true if we were forced in D-T systems to go to solid breeders for some reason or another.

The lack of tritium in the blanket can have several beneficial effects, one of which is to lower the overall tritium inventory in the reactor. Unfortunately, we have seen previously that the tritium inventory in the blanket in only 10 to 20 per cent of the overall value in low burn-up devices such as TOKAMAKs, and elimination of this amount will not

drastically alter the potential hazard of a fusion reactor. The effect would be much larger in high burn-up devices such as inertial confinement reactors.

The production of tritium by the D-D reaction means that there will still be considerable tritium in the plasma. D. Steiner [A-1] shows that for a mirror device operating at approximately 300 keV, there is roughly 30 per cent as much T_2 in the D-D plasma as in a D-T cycle. This of course assumes that one would recycle and "burn-up" the T_2, thus increasing the efficiency. But if one chose to simply store the T_2 from the plasma exhaust, the inventory would build up very fast. Steiner's calculations indicate that as much as 100 kg of T_2 would be produced per year of operation of a 3000 MW(th) D-D fusion reactor. This approach would largely negate the advantage of no *required* T_2 production, and would seem counter-productive. Assuming that one recycles the T_2, then the inventory in the fuel cycle (which would now be the total inventory, because no T_2 is *required* for storage) would be reduced to roughly one-third of the normal D-T value. This would translate into approximately 1 kg/1000 MW(th), which is roughly 10 per cent of the inventory in "normal" D-T systems.

A very important side effect of not having T_2 in the blanket is that the routine releases would be considerably reduced. The major pathway for T_2 release to the environment in D-T systems is through the power cycle, especially if it contains water. Removal of the tritium from that high temperature part of a fusion reactor would certainly be a step in the right direction.

Finally, there is considerable uncertainty at the present time about the lowest practical T_2 concentration that could be achieved in a liquid or solid breeder for a D-T system and still allow economical extraction procedures. Values of around 1 appm are assumed without much experimental evidence, and it is possible that values ten times this may be more appropriate. Hence, if there was no need to breed tritium, there would be no need to develop a *very* expensive technology associated with such chemical separations and the redundant containment structures that would almost certainly be required.

Another advantage of the D-D fusion cycle is that a much wider range of coolants and structural material could be used. One would probably *not* use Li-containing coolants, because to do so would negate the lower tritium inventory advantage. However, since neutron economy is no longer of primary concern in the D-D cycle, coolants such as water, sodium, organics, fluidized beds or even molten salts could be used. The structural materials can now be chosen a bit more freely, not subject to the constraints of Li corrosion, parasitic neutron absorption, or compatibility with tritium. Alloys of zirconium, titanium, aluminum, nickel, or magnesium could be considered. Such choices would, of course, have to be compatible with the thermal and mechanical loads, neutron and charged particle damage, as well as the new coolants.

In contrast to the popular conception that neutron-induced radiation damage would be less in a D-D system, recent studies have shown this not to be the case. Before discussing the reason for this we must speak about the mechanisms one can use to compare a D-D cycle to a D-T cycle.

The "normal" method of comparison is on the basis of neutron energy passing through the first wall (i.e., MW/m^2). This "neutronic" definition for the two systems reveals the following neutron fluxes:

$$\text{D-T:}\quad 1\ \frac{MW}{m^2} = \left(14.1\ \frac{MeV}{n}\right)\cdot\left(4.43\cdot 10^{17}\ \frac{n}{m^2\cdot sec}\right)\cdot \left(1.602\cdot 10^{-19}\ \frac{MW\cdot sec}{MeV}\right)\ ;$$

$$\text{D-D:}\quad 1\ \frac{MW^*}{m^2} = \left(14.1 + 2.45\ \frac{MeV}{n}\right)\cdot\left(7.56\cdot 10^{17}\ \frac{n}{m^2\cdot sec}\right)\cdot \left(1.602\cdot 10^{-19}\ \frac{MW\cdot sec}{MeV}\right)\ .$$

This does not give a normalization on the basis of power generated. To accomplish thermal normalization, we must total up all the energy produced in the plasma. When this is done, we get the following neutron fluxes:

$$\text{D-T:}\quad 1\ \frac{MW}{m^2}\left(\frac{\text{Plasma}}{\text{thermal}}\right) = \left(14.1 + 3.5\ \frac{MeV}{n}\right)\cdot\left(3.55\cdot 10^{17}\ \frac{n}{m^2\cdot sec}\right)\cdot \left(1.602\cdot 10^{-19}\ \frac{MW\cdot sec}{MeV}\right)\ ,$$

$$\text{D-D:}\quad 1\ \frac{MW}{m^2}\left(\frac{\text{Plasma}}{\text{thermal}}\right) = \left(\frac{4.04+3.27+17.6+0.2(3.67+14.67)}{2}\ \frac{MeV}{n}\right)\cdot \left(4.3\cdot 10^{17}\ \frac{n}{m^2\cdot sec}\right)\cdot\left(1.602\cdot 10^{-19}\ \frac{MW\cdot sec}{MeV}\right)$$

---

* assuming all the T_2 is burned.

We think that a comparison of displacement damage, helium production, and induced radioactivity, based on the *plasma-thermal* normalization is most appropriate, and we will proceed to discuss the problem in this light.

The ratios of these critical radiation-damage indices in a D-D system to those in a D-T system are given in Table A-I. The striking point about the results in Table A-I is that the displacement damage is hardly effected at all in an equal thermal wall loading, while the helium production is only reduced by roughly 40 per cent. (The reason why the reduction is constant is that the helium-production threshold energy for all the elements considered is well above 2.45 MeV, and the reduction is in proportion to the reduction of the 14.1 MeV neutron flux.) The radioactivity is reduced by factors of 15 to 70 per cent, with V being most sensitive.

It is on the basis of these results that we could contend that there is little to be gained from a *radiation-damage and induced-radioactivity standpoint* by moving to the D-D fuel cycle.

The main disadvantage of the D-D cycle is the higher operating temperature required. Not only is this a question of technology which has not been developed, but it translates into higher magnetic field or laser requirements than for a D-T system. The capital cost of such advanced systems is bound to be higher than for "present" D-T devices, and it is uncertain exactly how much of a cost penalty would have to be paid (assuming such plasmas can be produced at all).

Table A-I: Comparison of Relative Displacements, Helium Gas Production Rates and Induced Radioactivities in D-D versus D-T Fuel Cycles (on the basis of 1 MW/m^2 plasma thermal wall loading) [A-2] (Ratio of (D-D/D-T) Effects)

Element	Displacements	Helium Gas Production	Induced Radioactivity* t = 0
Al	1.12	0.61	0.86
V	1.03	0.61	0.35
SS 316	1.07	0.61	0.78
Nb	0.91	0.61	0.54
Mo	0.93	0.61	Not available
Ta	0.90	0.61	0.68

* After 2 years of operation.

Finally, the lower energy content per reaction translates into lower power densities in the plasma, which in turn means that larger magnetic systems would have to be built for a D-D fuel cycle. Presumably this would not apply to laser reactors. Larger systems mean higher capital costs, and again higher costs of generating electricity. Unfortunately, we cannot give quantitative numbers for this effect at the present time.

In summary, the major advantages of a move to a D-D cycle would be the reduction (but not elimination) of the tritium inventory and routine T_2 releases to the environment. A much wider choice of coolants and structure could ease the materials resource problem, and the fact that no lithium would be required would mean that an essentially infinite fuel supply was available.

There seems to be little advantage from a radiation-damage or induced-radioactivity standpoint. The lower power density and the higher operating temperature would most certainly increase the cost of generating electricity. The fact that we have not yet been able to control plasmas at one-tenth of the temperatures required for the D-D systems means that it will certainly be a long time in the future before we could utilize this fuel cycle.

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APPENDIX B

FUSION-FISSION HYBRID REACTORS

J.P. Holdren

Fusion-fission hybrids are systems wherein a fusion core is surrounded by a fission blanket. The fusion core may be either a magnetically confined plasma or an inertial-confinement system, wherein compression and heating of the fusion fuel is provided by lasers, electron beams, or ion beams. The fission blanket may be designed to exploit fast fission, thermal fission, production of fissile isotopes from fertile ones, or a combination of these.

Hybrid systems have been receiving increasing attention in the world nuclear community, particularly in the United States and the Soviet Union [B-1 to B-6], in connection with three quite distinct applications: *(a)* use of hybrids as self-contained, base-loaded electricity generating stations; *(b)* use of hybrids primarily as producers of fission fuels for use in separate pure-fission reactors; *(c)* use of hybrids for the transmutation of long-lived radioactive wastes, primarily by fast fission of the actinides.

1. BASE-LOAD ELECTRICITY GENERATION

In the self-contained, electricity-generation application, the primary function of the fission blanket is to multiply the energy yield obtainable from each 14 MeV fusion neutron. Increasing the energy yield per neutron opens the possibility of significant net energy production in a device whose fusion core by itself would be a marginal energy producer, or even a net energy sink. In this role of the hybrid, then, the idea is that the fission blanket enables fusion technology to find useful application even before an economic pure-fusion system becomes available. Since energy multiplications of about a factor of ten appear to be achievable in conceptual fission blankets that have been analyzed, the requirement on the Lawson parameter, $n\tau$, where n is the fuel number density and τ the confinement time in the fusion plasma, can be relaxed by about a factor of ten below what would be required in a pure fusion system.

The relaxation of the Lawson requirement exacts a significant price in technical complexity (hence presumably construction costs), and in the loss of potential environmental advantages of pure fusion. With respect to complexity, meeting the elaborate requirements of fusion and fission subsystems (injection, heating, confinement, and tritium breeding for fusion; refueling for fission; cooling and heat removal for both), in the close proximity required by the neutronics, poses formidable difficulties which are unlikely to be overcome cheaply. (For a specific conceptual design, see, e.g., [B-7].)

With respect to environment, reliance on fission for the bulk of the energy production means that the hybrid reactor's radioactivity inventory, decay-heat, fissile-material production, and long-lived radioactive wastes are essentially those of a fission system. But hybrids might have a significant safety advantage over some pure fission reactors in terms of accident pathways: specifically, a hybrid's fission blanket will be subcritical in its operating configuration, and possibly can be designed also to be subcritical under all accident conditions; also, the power density in the fission blanket may be made low enough to substantially alleviate emergency-cooling requirements. Neither of these potential advantages is automatic [B-8]. In some designs, a blanket that is subcritical in normal operation could become critical in accidents, involving loss of coolant and severe geometric reconfiguration. Achieving energy multiplication that is high enough to make a power-producing hybrid really interesting may require a high power density in the blanket.

2. HYBRIDS AS FUEL PRODUCERS

In this application, the primary function of the fission blanket is to produce fuel for use in physically separate pure-fission reactors. The breeding reactions of interest are U238 → Pu239, and Th232 → U233. Although many designs would produce both power and fuel, it is not essential that a hybrid intended for the fuel-production mode also produces net electrical power; a negative net electrical-power balance in the hybrid could be made up from the pure-fission power grid to which the hybrid was supplying fuel. This is what is meant by a "symbiotic" relation between hybrids and pure-fission reactors.

Even if hybrid technology proved too expensive to be attractive in the self-contained, electricity-generation role, it might well be affordable in the fuel-producing role. This is so because one large fuel-producing hybrid could supply five to ten fission converter reactors (LWRs or HTGRs, for example) with make-up fuel. Alternatively, fuel-producing hybrids might be used in a fission-breeder reactor economy to reduce the effective doubling time, and thus increase the rate at which installed generating capacity in breeders could be expanded. The fuel-producing role of hybrids also has the advantage that hour-to-hour and day-to-day reliability is less critical than for an

electricity-producing reactor tied directly into the power grid. This is an especially important consideration for a new technology that might be expected to have unpredictable reliability, while bugs are being worked out.

Hybrids optimized for the fuel-producing role (with fissions in the blanket deliberately suppressed) can have somewhat smaller inventories of fission products and lower blanket-power densities than electricity-producing hybrids. Safety ramifications must be analyzed in a systems sense, however. Two possible "systems" safety ramifications of fuel-producing hybrids are: *(a)* hybrids could permit the pure-fission reactors in the system to be HTGRs or heavy-water CANDUs (for example) instead of LMFBRs, or low-breeding ratio LMFBRs instead of high-breeding-ratio ones, which would be important if certain converter reactors or low-gain breeders were deemed safer than high-gain breeders; *(b)* hybrids might permit the use of the Th232 → U233 fuel cycle where otherwise U238 → Pu239 was required, which would be important if the thorium cycle were deemed safer than the plutonium cycle (for example, in respect to safeguards against production of nuclear explosives).

3. HYBRIDS FOR TRANSMUTATION OF RADIOACTIVE WASTES

The most dangerous long-lived radioactive wastes from fission fall principally into two categories: fission products of medium weight and half-life near 30 years (dominated by Sr90 and Cs137), and heavy elements (the so-called actinides) formed by absorption of neutrons in fuel isotopes and by the subsequent decay chains. (Among the most troublesome actinides are Am246 [half-life: 458 yr], Am243 [half-life: 7950 yr], Pu239 [half-life: 24,000 yr], and Np237 [half-life: 2 million yr].) The possibility of reducing the longevity of the hazard of fission products and actinides by bombarding them with fusion neutrons has now been studied in some detail [B-5, and B-9].

The absorption of fusion neutrons by Sr90 and Cs137 produces isotopes of somewhat shorter half-life and/or lower radiological toxicity, but the neutron fluxes needed to achieve these transmutations at a useful rate are formidably high. Even if a hybrid could be designed to produce such neutron fluxes and to withstand the associated materials damage, the long exposure times required to transmute the fission products imply the accumulation of very large fission-product inventories in the hybrids themselves (larger than in pure-fission power reactors).

More promising is the use of fusion neutrons for fissioning the long-lived actinides, thereby converting them into still very toxic but substantially shorter-lived fission products. This process could lead to useful reductions in overall hazard lifetime at neutron fluxes that seem likely to be achievable

(around one MW of neutron power per m^2). A hybrid reactor designed for this purpose would contain at any given time a very high inventory of dangerous actinides, however, and accident pathways for such hybrids will have to be scrutinized very carefully to determine whether this scheme is desirable. Another factor in such a determination is the extra handling and shipment of actinides that might be involved in comparison to alternative waste-management schemes. Since actinides can also be "burned-up" in place in fast-neutron pure-fission reactors (with a loss of breeding of new fissile fuel), it might be preferable to do that, and make up the loss of breeding by using hybrids in their fuel-producing role [B-5].

4. CONCLUDING OBSERVATIONS

The rationale for developing hybrids (or any other new energy source) presumably consists in potential advantages over alternative systems with respect to one or more of the following characteristics: abundance of fuel, cost of delivered energy, timing, and environmental and social characteristics. The LMFBR already solves the fuel-abundance problem for many thousands of years, and pure-fusion reactors, if practicable, will solve it for even longer, so there is little advantage in this respect for fusion-fission hybrids. With respect to cost of delivered energy, it is also hard to make a compelling argument for fusion-fission: the capital costs of hybrids, which will dominate the energy cost, are quite uncertain but likely, for basic engineering reasons, to be higher than those of pure fission or pure fusion. With respect to timing, hybrids have the potential advantage over fusion of being deployable *sooner*, and the potential advantage over LMFBRs of permitting a *more rapid expansion* of nuclear capacity (owing to a higher breeding ratio). The possible early availability of hybrids and the related "stepping stone" argument (that learning from hybrids will bring about pure fusion sooner) are rather weak rationales for hybrid development, unless hybrids are better than alternative "interim" technologies in other respects. The hybrid's advantage for rapid expansion of nuclear capacity is weakened somewhat by the increasing likelihood of slower electricity growth in the industrial nations, where the main market for this sophisticated technology lies, and, in some countries, by the accumulating inventory of plutonium from converter reactors, which makes the low breeding ratio of the LMFBR less of a liability.

To the extent that the rationale for hybrids on grounds of fuel supply, cost, and timing is perceived as marginal, the environmental and social characteristics in comparison to alternative technologies take on increased importance. In evaluating environmental as well as other characteristics of hybrids, it is essential to determine the appropriate "yardsticks" against which hybrids should be compared. The answer depends on which of the possible roles of hybrids is under discussion. In the electricity-production role, the appropriate comparisons are

with pure-fission and pure-fusion reactors (and, in a broader context, with non-nuclear generating technologies); in the fuel-production role, the comparison is with mining and enrichment or with pure-fission breeders; and in the waste-transmutation role the comparison is with other waste-management schemes.

Characteristics of hybrids that are relevant to evaluating environmental risks in any such comparison are: *(a)* radioactive inventories, including tritium, activation products, fission products, and actinides (including fissile isotopes); *(b)* pathways for release of the inventories, relevant to which are criticality behavior, response to loss of coolant or coolant flow; other stored energy forms; "geometrical" aspects (seams, welds, valves, lengths of pipes); and the amount of transport and handling of radioactive materials that is required; *(c)* systems aspects, including the impact of hybrid technology on fuel choices and fission-reactor mix within the nuclear system as a whole. Detailed work on these characteristics is still largely missing. It should be undertaken without delay if fusion-fission hybrids are to be taken seriously as an additional option for the medium-term and long-term phases of the energy future.

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APPENDIX C

METHODS OF COST EVALUATION IN THE USSR

It is outlined in the Introduction that the study is not meant to concentrate on costs, and capital costs in particular. Realizing, however, how controversial this topic is, we have taken advantage of the cooperation of our Soviet colleagues and asked them to provide their methods of evaluating capital costs of future technological projects.

C-I DETERMINATION OF ECONOMIC EFFICIENCY OF CAPITAL INVESTMENTS IN POWER PROJECTS OF THE USSR

(M.A. Styrikovich, A.M. Belostotsky)

The economic efficiency of capital investments in power projects is determined in the process of planning, at the stage of study, when contract designs are worked out and the construction of these projects is substantiated in technical and economic terms. These economic calculations are part of a multi-stage iterative process of planning the USSR national economy; this is effected by branch ministries and departments under the methodological and practical supervision of the USSR State Planning Committee, the chief planning body of the country. In the planning process, the indices of the national economy's fuel and power requirements are determined and coordinated, as well as the general economic estimates of the material and other resources and services of non-power branches which are used for developing the power industry.

Calculations for determining the comparative economic efficiency of capital investments in the power industry serve to compare variants for economic and technical solutions, the location and construction of new enterprises, or the reconstruction of operating ones. This also involves the selection of their parameters, as well as the comparative evaluation of the efficiency of new machinery and equipment.

The methods of calculation are developed at the branch level; they are endorsed by the USSR State Planning Committee.

1. COMPARATIVE ECONOMIC EFFICIENCY OF CAPITAL INVESTMENTS

The index of comparative economic efficiency of capital investments is the so-called calculated costs determined by formula

$$C = E_k K + O \quad , \tag{1}$$

where

- C = calculated costs for one of the variants under study;
- O = annual operating and maintenance expenses;
- K = total capital investments for the project; and
- E_k = standard coefficient of efficiency of capital investments.

For an accurate comparison of the variants it is necessary to use similar quantities and qualities of what is produced.

A variant is considered most efficient if its value of calculated costs is the least. The indices K and O can be used both fully and as specific values.

E_k used in formula (1) is determined by optimization or evaluation of the national economy, and is pre-set for a given branch. The economic essence of this index lies in that it characterizes the efficiency of the last project to be included in the optimum national economic plan. Another interpretation of this index is obtained by comparing two variants of project construction. Let us assume that the first variant is characterized by smaller capital investments ($K_1 < K_2$), but greater operating and maintenance expenses ($O_1 > O_2$). The index E_k is used to commensurate capital investments and operating and maintenance costs, so that

$$C_1 = E_k K_1 + O_1 \quad , \qquad C_2 = E_k K_2 + O_2 \quad .$$

Supposing the second project can save the operating and maintenance costs to a sum $O_1 - O_2 = \Delta O$, then the permissible overdraft of capital investments is determined when $C_1 = C_2$ with

$$\Delta K = K_2 - K_1 = \frac{\Delta O}{E_k} \quad .$$

Having expressed from here E_k, we get

$$E_k = \frac{\Delta O}{\Delta K} \quad .$$

If we take

$$\frac{1}{E_k} = T \quad ,$$

then

$$T = \frac{1}{E_k} = \frac{\Delta K}{\Delta O} \quad ,$$

and E_k can be interpreted as the value inverse to the compensation period for additional capital investments.

E_k is periodically revised (approximately once in every five years). At present $E_k = 0.12$ is in operation.

2. DETERMINING CALCULATED COSTS

The value of calculated costs is determined according to formula (1), provided capital investments are made within one year, while annual production costs remain constant for the entire period of operation.

In actual fact, capital investments are spread over all the years of the construction period, and the project under construction is temporarily commissioned when annual production costs change with time.

In this case the comparison of variants is made by reducing the outlay to one year* by using the reduction coefficient

$$B = \frac{1}{(1+E_d)^t} \quad ,$$

where

* It is evident that, although the choice of the years to which the outlay is reduced influences the value of the latter, it does not influence the relative efficiency of the variants compared.

B = reduction coefficient;

t = period of time in years;

E_d = norm for calculating the outlay made at different times. Just as the norm of efficiency, E_d is pre-set, and E_d operating at the present time equals 0.08*.

If the deadlines of construction and the annual productivity changes of the project are not taken into account, the specific calculated costs can be fixed by formula

$$\bar{C} = \frac{E_k K + O}{W} ,$$

where W is the volume of annual output invariable in time (or capacity).

If the volume of annual output changes and the distribution in time of construction deadlines are taken into account, the specific calculated costs can be determined by the formula

$$\bar{C} = \frac{\sum_{t=1}^{T} (E_k K_t + O_t)(1+E_d)^{\tau-t}}{\sum_{t=1}^{T} W_t (1+E_d)^{\tau-t}} .$$

The last expression is obtained from the following:

The specific calculated costs are the mean value of the index in time, which, if multiplied by the output produced, gives the costs corresponding to capital investments and annual costs over time.

Indeed, by assuming the specific calculated costs constant (mean), the total costs will be as follows:

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* Provided the depreciation charges are determined without discount, that is % assigned for renovation equals $100/\tau_e$, where τ_e is the calculated period of the equipment life. Calculation can be made with growing deductions by formula of compound interest, with reduction coefficients B. In this case $E_d = 0.12$.

$$\sum_{t=1}^{T} (\bar{C}\ W_t)\,(1+E_d)^{\tau-t} \quad .$$

Equating them with the real calculated costs we get

$$\sum_{t=1}^{T} (\bar{C}\ W_t)\,(1+E_d)^{\tau-1} = \sum_{t=1}^{T} (E_k K_t + O_t)\,(1+E_d)^{\tau-1} \quad .$$

From here we can directly obtain the expression for $\bar{C}$.

Table C-I gives an example of adjusting specific capital investments, accounting for the variable volume of output and the unevenness of capital investments in various years of construction.

Table C-I: Determination of Specific Reduced Capital Investments

Indices	Years of Construction and Commissioning of Capacity						
	1	2	3	4	5	6	Total
Reduction Coefficient $(1 + E_d)^{\tau-t}$	1.47	1.36	1.26	1.17	1.08	1.00	-
Capital Investments in Power Plant K_t %	8	14	18	22	23	15	100
Calculated Capital Investments in Power Plant $K_t(1 + E_d)^{\tau-t}$ %	11.8	19	22.7	25.7	24.8	15.0	119
Increases in Commissioning of Capacities ΔN_t %	-	-	25	25	25	25	100
Reduced Increases in Commissioning of Capacities %	-	-	31.5	29.2	27.0	25.0	112

Note: Calculation is made to the last year, i.e. $\tau = 6$.

The period of construction and the distribution of capital investments and increases in capacities over years are determined in accordance with the existing practice and are fixed in norms.

In this example the ratio between reduced capital investments and reduced capacity increases is 119/112.7 = 1.06, i.e. 6 per cent more than compared with the variant without discounting.

This ratio is always somewhat greater than unity, because capital investments should be made with some advance with regard to the commissioning of projects, but it is different for different projects because the distribution of capital investments and the commissioning of capacities by years do not coincide.

3. COMPARISON OF VARIANTS

The simplest method of reducing variants so that they become comparable is to use specific indices (per kW power, kWh of supplied energy, etc.). An essential shortcoming of such a comparison is that it neglects an important factor--the size of enterprises, which leads to noticeable errors if sizes differ considerably.

Nevertheless, this method can be useful in evaluating new kinds of machinery and equipment, when design estimates are not yet ready and the project under study is not timed to a concrete load, region, scale of power consumption, etc.

In designing power industry projects other methods are preferred. There is a possibility of examining more thoroughly the commissioning of capacities and power production (for electric power stations) over the years of the calculation period. Another method is the conventional addition of corresponding capacities and power according to economic indices typical of the region, power system, calculated period, etc. under study.

4. COMPOSITION AND STRUCTURE OF CAPITAL INVESTMENTS

Capital investments K take into account the outlay for the building of enterprises, the cost of equipment and its assembly, expenditures for geological and other prospecting and design and preparatory work involved in the construction of a given enterprise. The composition and structure of capital investments can be exemplified by the design of a coal-fired generating power plant with a capacity of 2400 MW and three 800 MW units. The specific capital investments in this enterprise amount to 135 Rub/kW. The distribution of the outlay is shown in Table C-II.

Table C-II: Structure of Capital Investments in Power Station Construction

Items of Outlay	Item Outlay Ratio to Total Sum (%)
1. Construction and Assembly	
Materials and semi-finished products	26
Man-power expenditures	6
Expenditures on exploitation of construction machines and equipment	4
Other outlay	2.8
Overhead expenses	7.2
Profit of construction firms	2.9
Subtotal 1	48.9
2. Cost of Production Equipment	33.5
3. Cost of Other Equipment	10.3
4. Other Expenses:	
Adjustment, finishing, tests	0.7
Business trips	0.5
Maintenance of construction management	0.3
Design and prospecting work	1.1
Other (including unforeseen expenses)	4.7
Subtotal 4	7.3
Total	100

Outlay for the equipment is represented by: outlay for production equipment, and outlay for other equipment. The latter includes electrical equipment, testing instruments, and automatic equipment.

The structure of the outlay for production equipment is given in Table C-III.

Table C-III: Structure of Outlay for Production Equipment (%)

Items of Expenditures for Equipment	Item Outlay Ratio to Sum of Expenditures for Production Equipment	Expenditures for Maintenance of Power Station
1. Boiler section,	37	12.4
including boiler unit	25.8	8.6
2. Engine room,	39	13
including turbines with condensers	27	9.2
3. Pipes and fittings	6.4	2.2
4. General equipment for power station	2.8	0.9
Total: Main building	85.2	28.5
5. Outer installations	14.8	5
including fuel preparation and supply	8.6	2.9
Total: Power station	100	33.5

5. OPERATION AND MAINTENANCE OF POWER PROJECTS

Operation and maintenance of power projects are determined according to the formula

$$O = O_p + O_f + O_d + O_e \quad ,$$

where

O_p = permanent annual production costs (that do not depend on the volume of output);

O_f = expenditures for fuel (variable annual production costs);

O_d = sum of depreciation costs (for projected new equipment installation costs and capital repairs); and

O_e = operational expenditures on exploitation, including those for routine maintenance, wages, expenses for general purposes, and others.

The values of specific constant outlay are determined by norms set by ministries. For example, for the coal-fired power plant mentioned above, the following indices should be accepted, in accordance with the norms:

(a) depreciation costs (7 per cent) for capital investments in power plant*:

$$135 \text{ Rub/kW} \cdot 0.07 = 9.4 \text{ Rub/kW} \quad ;$$

(b) routine maintenance costs (18 per cent of depreciation costs):

$$9.4 \text{ Rub/kW} \cdot 0.18 = 1.7 \text{ Rub/kW} \quad ;$$

(c) expenditures on wages (man-power factor for coal-fired power plant is 0.37 man/MW, with average annual wages of 1500 rubles per person):

$$1500 \text{ Rub/man} \cdot 0.37 \text{ man/MW} \cdot 10^{-3} = 0.6 \text{ Rub/kW} \quad ;$$

and

(d) general outlay and other expenses (27 per cent of the sum of depreciation costs, routine maintenance and wages):

$$(9.4 + 1.7 + 0.6) \cdot 0.27 = 3.1 \text{ Rub/kW} \quad .$$

The constant part of annual production costs thus equals

$$O_p = 9.4 + 1.7 + 0.6 + 3.1 = 14.8 \text{ Rub/kW} \quad .$$

Annual outlay connected with fuel expenditures is calculated on the basis of

(a) planned load factor of power station (the base load adopted is h = 6500 h/yr);

Including 3.3 per cent for projected new equipment installation costs (taking life time T_i to be 30 years) without taking discounting into consideration, and 3.7 per cent for capital repairs.

(b) specific fuel consumption corresponding to these operating conditions (b = 340 g/kWh); and

(c) cost of fuel based on the results of optimization of fuel and power balance (by shadow prices*), for the calculation it is assumed that p = 25 Rub/trf (rubles per ton of reference fuel).

Thus, the annual outlay for fuel amounts to

$$O_f = 340 \text{ g/kWh} \cdot 6500 \text{ h/yr} \cdot 25 \text{ Rub/trf} \cdot 10^{-6}$$

$$= 55.2 \text{ Rub/kW} .$$

6. SUMMARY

Summing up the components of outlay obtained after calculations in accordance with formula (1) we get

$$C = E_k K + O_p + O_f = 0.12 \cdot 135 + 14.8 + 55.2$$

$$= 16.2 + 14.8 + 55.2$$

$$= 86.2 \text{ Rub/kW} .$$

The calculated costs of coal-fired power plants can be compared with other variants of the power supply. But the equality of variants is obligatory as far as output and capacity are concerned.

The indicated values of expenditures can serve as a reference for evaluating efficient new machinery and equipment in the power industry.

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* The economic essence, formation,and methods of calculating shadow prices are given in Appendix C-III.

C-II EVALUATION OF THE MAXIMUM PERMISSIBLE COST INDICES OF FUSION POWER PLANTS IN THE USSR*

(N.N. Vasiliev)

When fusion power plants will be developed at a commercial scale and become well-advanced generating units, their economic efficiency will have to be compared to that of future power plants of other types to be installed in the same period [C-3].

Naturally, such a comparison must contain many assumptions, since one must account not only for the progress of fusion-power-plant-systems but also for the advancement of nuclear (fissile) power plants as well as the development of conventional fossil power stations in the USSR. The latter may use not only costly fuels but also extremely cheap coal like that which is strip-mined at Kansko Achinsk in Siberia.

It is quite obvious that there are also many assumptions that are associated with the uncertainty of forecasting economic indices for both power plants and the national fuel and power industry as a whole. These assumptions must not be separated from possible differences between hypotheses on the development of the national economy. Major trends of this development are subject to the well-known laws of countries with planned economies. However, the predicted parameters may vary quantitatively. It goes without saying that all the estimates must be given in today's prices.

The initial values required for defining the permissible capital investments per MW(e) of installed fusion-power- plant capacity must include:

(a) The maximum specific cost of one generated MW(e)·h, taking into consideration possible deviations of the electrical energy balance; and

(b) Estimates of the achieved specific generating costs for producing one MW(e)·h at different types of power plants, taking into account that the economic indices depend on the differences in capital investments of those power plants.

The structure of the calculated electric·energy production consists of the fuel-cost constituent, and a component that mainly depends on capital investment (measured in Rub/MW(e)·h):

$$C = O_f + c_c \quad , \tag{2}$$

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* Appendix C-I defines terms and the standard method of calculating the economic efficiency of capital investments in energy objects in the USSR.

where O_f is the fuel-cost component of the specific cost of electric energy production, and the cost component $c_c = f(K_{sp})$ is a function of the specific capital investments K_{sp}.

It should be noted that in equation (2) K_{sp} may be greatly influenced by the load factor (or: annual operation factor). However, all our reasoning refers to power plants working at base load. Therefore, the dependence of c_c on the factors associated with the operation mode of the plant may be neglected.

In general, if the price of fossil fuel is up, O_f will also rise and, in spite of the fact that fossil power plants have relatively low capital investments as compared to nuclear power plants, this will result in an increase in energy production cost (C). The fuel cost component of nuclear power plants tends to decrease. Thus, their competitiveness vis a vis fossil power plants is assured, even when the specific capital costs (K_{sp}) of nuclear power plants increase.

In turn, the maximum permissible value of the specific calculated costs (C) of electric energy production, measured in Rub/MW(e)·h, depends upon the C that is actually achieved. Therefore, if one assumes hypothetically that electricity production by nuclear power plants in the overall national energy output will be small, then one concludes that C may increase as the result of rising fuel prices, i.e. of O_f. Then, assuming C to be high, it is possible to suppose that fusion power plants, whose fuel costs O_f are very low, could have considerable specific capital investments K_{sp} and, correspondingly, a big cost component c_c. However, such a supposition may be critized for the two following reasons.

First, it is incorrect to assume a growth rate of the maximum permissible, specific calculated electricity production costs that is too high, since this would imply a distortion of the proportions of the national economy development. And this could mean higher prices of industrial goods and services, or a limitation upon the level of industrialization and communal electrification.

Second, one should bear in mind the objective trends of technological progress. These trends have caused a substantial rise of the share of nuclear power plants in the national energy balance, and have improved their economic indices. These latter features will determine the stability of C or influence its relatively small increase. Such an assertion is true even if both fossil and nuclear fuel prices rise, and if more expensive fast-neutron reactors are to be introduced to satisfy the growing fuel requirements of nuclear power engineering.

Around 1995 to 2000, fusion power plants should be compared to the most highly developed nuclear power plants which, by that time, will be greatly advanced. Let us proceed from the hypothesis that nuclear power plants will substantially prevail

among new power capacities in future electrical grids. In this case the economic indices of the most expensive nuclear power plants will be close to the maximum permissible value of the calculated specific cost of electricity production C_0. In other words, $C_{NPP}^* \leq C_0$, with the asterisk indicating the largest capital investments. The value of C_{NPP}^* may be defined as the calculated specific electricity cost of generating one MW(e)·h at a nuclear power plant with a fast-neutron reactor. Thus C_{NPP}^* could be calculated assuming the initial capital investments to be up to 400 Rub/kW(e).

On the other hand, as was pointed out above, by that time fossil power plants operated on cheap Kansko-Achinsk coal would be largely developed. It is worth mentioning that the value C_{CPP}^* (electricity-generation cost of coal power plants) is much less than that of C_{NPP}^*. According to the author's first approximations [C-3], C_{NPP}^* equals 12.5 to 13.0 Rub/MW(e)·h, compared to C_{CPP}^* equaling 5.5 to 6.0 Rub/MW(e)·h for a conventional fossil power plant fueled with Kansko-Achinsk coal.

On the basis of the above calculations, it was decided to assume the maximum-permissible value of specific electricity-production cost to be greater or equal to 13 Rub/MW(e)·h (this value refers to the national electricity production system as a whole). It means that with

$$C_{FPP} = C_0$$

and using formula

$$c_c = 0.21 \frac{K_{sp}}{(6.5 \text{ to } 7.0) \cdot 10^3} ,$$

the maximum permissible specific capital investments for fusion power plants will be:

$$K_{sp} = (400 - 430) \cdot 10^3 \text{ Rub/MW(e)} .$$

As to the first large commercial fusion power plants, their permissible capital investment may be considered to be within the range of 440 to 500 Rub/kW(e).

These capital costs may be used as a certain criterion for the evaluation of various concepts of fusion power plants.

We do not think that nowadays we know absolutely everything about the processes in fusion reactors and their operating characteristics, including safety and reliability features. That is why, to some extent, any cost estimates will be approximative.

From these general considerations it becomes clear that an increase in efficiency with regard to safety requirement limitations may be attained by

- raising specific reactor parameters (e.g. the energy loading on the first wall);
- increasing the efficiency of the conversion of thermal energy to electrical power; and
- reducing the tritium inventory quantity in fusion power plants.

C-III CALCULATIONS OF TECHNICAL AND ECONOMIC FACTORS IN FUSION POWER STATIONS (PROJECTS TVE-2500 AND LTB-500)

(N.N. Vasiliev, R.R. Grigoriants)

The basic technical and economic factor of any power plant, including fusion power plants, is the quantity of calculated costs C, which, in the end, is *the* criterion for comparing the commercial efficiency of plants with different technologies of energy production.

The calculation method used in the USSR to determine the quantity of expenditures is described in Appendix C; therefore, no additional explanations are given for the calculation formula introduced here.

The calculated costs include two components: capital investments, and annual operating and maintenance expenses:

$$C = E_k K + O \quad , \tag{3}$$

where:

E_k = standard coefficient of efficiency of capital investments;

K = total capital investments for the project; and

O = annual operating and maintenance expenses.

In calculating factors K and O for fusion power stations, the following basic facts must be taken into account:

- The station is a serial power station; and
- Expenses for scientific research are not included in the estimated cost of the object.

1. Calculation of Capital Investments K

The quantity of capital investments includes the costs of equipment, its assembly, buildings, technological and auxiliary systems. The equipment of the fusion power stations TVE-2500 and LTB-500 consists of: standard equipment; planned equipment for future utilization; and completely new equipment. Accordingly, for calculations of fusion power stations with a TOKAMAK reactor and a magneto-hydrodynamic energy conversion system (MHD) (project TVE-2500), the following classification of equipment was used:

Group "A": Standard equipment or equipment of which industrial analogues are available today;
This group includes steam turbines with condensers and regenerative heat exchangers, a feedwater preparation system, a feedwater recycling system, etc., a system of water supply including cooling towers; different types of auxiliary equipment; generators, etc.

Group "B": Equipment applying most recent technologies, which is beyond the experimental stage but still without industrial analogues.
This group includes MHD generators with auxiliary equipment, but without magnetic system (which belongs to *group "C"*); steam generators with high-temperature gas heat carriers; compressors; a number of technical systems of the main building.

Group "C": New equipment applying technologies which have not been studied very much. This group includes the fusion reactor proper, injectors, the system of fuel preparation and fuel intake, the system of tritium separation and tritium regeneration, etc.

The approach towards a cost estimate for the equipment and its assembly must be different for *groups "A", "B",* and *"C"*.

The cost of equipment of *group "A"* can be determined directly with the help of valid lists (prices must be given for the year when the project was worked out).

The cost of elements of *group "B"* is determined on the basis of detailed project studies that take into account the results obtained when industrial or experimental industrial analogues are built and utilized.

The calculation of the cost of a steam generator or a TVE-2500 plant may serve as an example.

Construction calculation studies of the steam generator help to determine its basic technical characteristics: the size of heat transfer surfaces, diameters and thickness of pipe walls, the nomenclature of materials to be used, etc.

The boiler aggregate was selected as industrial analogue of the steam generator. It operates within the reactor block with a serial steam-turbine plant with supercritical steam parameters at a capacity of 500 MW(e) on the turbine shaft.

The calculation was carried out with due regard to the constructional differences (the steam generator of the TVE-2500 plant does not have a number of elements which belong to the boiler aggregate-analogue), and the difference in material costs. Furthermore, if one takes into consideration that a higher reliability is required, a higher value of the coefficient of labor intensiveness (roughly 20 per cent) will be accepted.

Calculating capital investments for elements of the *"C" group* poses the biggest difficulty. The cost of this equipment must be calculated by using a number of assumptions, which are not very well-founded, with regard to tendencies of price changes for several new materials and equipment, and with regard to a substantial change in consumption. In this case one may use the method of maximum estimates to determine the upper cost level, so that the calculated costs for the power station in question do not exceed the maximum ceiling permitted.

However, direct cost estimates of *group "C"* are also necessary, for they will finally permit to calculate actual expenditures, given certain assumptions.

A further example, the approach towards the cost estimate of the superconducting magnetic system is described below.

The magnetic system of the TVE-2500 plant consists of different types of coils, power constructions, cryostats, electrical engineering equipment, and cryogenic equipment.

The specific cost of the magnetic system was accepted as a consolidated factor, and is to be determined in the following way;

$$\sigma_M = \frac{\sum_i K_i}{G}, \qquad \text{(Rub/kg)} \qquad (4)$$

where:

K_i = costs of individual elements of the magnetic system, with due regard for their production, assembly, and installation; and

G = total weight of superconducting windings (including the weight of stabilizing copper).

Electrodynamic and heat calculations as well as construction studies are carried out with regard to data on nuclear heat releases in superconducting windings. From these results one determines the basic construction parameters of the coils and cryostats, the material requirements and weight, energy, and dimensional data of the electrical engineering and cryogenic equipment.

The cost of refrigerators and electrical engineering equipment may be determined according to the method accepted for *group "B"* elements, although extrapolation is not so well founded here. For example, big oxygen aggregates may be accepted as analogue for helium refrigerators; in this case the calculation must be done according to specific indices, such as the cost of the aggregate per unit of weight and per unit of power to be consumed. The basic vagueness in this calculation is connected with the price level for a superconducting busbar. Thus, when changing over to large-scale production, the quantity of the consolidated factor σ_M may amount to roughly 30 to 60 Rub/kg.

The cost of one cubic meter of useful room in buildings was accepted as consolidated factor for evaluating their cost (taking into account hoisting and transport facilities, etc.), i.e. 60 Rub/m^3 for the reactor hall and other rooms with equipment using technologies that involve radiation hazards, and 40 Rub/m^3 for other rooms. The cost of cooling towers was included in the investments for the technical water-supply system.

The structure of capital investments for the construction of the fusion power station TVE-2500 is given in Table C-IV. The results shown reveal that, on given assumptions, the cost of the superconducting magnetic system amounts to 70 per cent of the overall capital investments.

2. Annual Operating and Maintenance Expenses

The second component of calculated costs are annual operating and maintenance expenses; they include expenditure in terms of fuel, maintenance, service, routine maintenance as well as total depreciation costs.

For the fusion power station TVE-2500 with a solid blanket, fuel costs consisted only of the costs of the deuterium to be burned, which are very small. The cost of the replaceable con-

Table C-IV: Structure of Capital Investments in the Construction of a Fusion Power Station: TVE-2500

	Total Investments (10^6 Rub)	Specific Investments (Rub/kW(e))
1. Main Building		
1.1 Cost of building	24	9.34
1.2 Reactor block (equipment and assembly)		
- Blanket	49	19.6
- Magnetic system (with cryogenic equipment)	785	274.3
- Injectors	15	5.84
- Other equipment	10	3.89
1.3 MHD generators (equipment and assembly)		
- Magnetic system	135	52.5
- MHD duct	7.9	3.07
- Other equipment	2.0	0.78
1.4 Thermo-mechanical block		
- Steam generators	16	6.23
- Turboaggregates	17.5	6.8
- Heat exchangers	2.4	0.92
- Other equipment	3.0	1.17
1.5 Technical systems of main building	50	19.45
2. Systems for the Entire Plant		
2.1 Transformer	22	8.56
2.2 Water supply system	30	11.67
2.3 Other equipment	12	4.67
3. Expenses Not Taken Into Account (10% of estimated cost)	110	42.9
Total	1212	472

Note: In calculating the capital investments for the construction of TVE-2500, the specific cost of the superconducting magnetic system (SCMS) was assumed to be σ_M = 50 Rub/kg.

struction elements with low work resources--the first wall and lithium-containing zones of the blanket--was taken into account when the overall depreciation costs were calculated (with the exception of the "first complex", the cost of which was included in capital investments). It was accepted that irradiated materials, among them lithium, are not regenerated.

The calculation of depreciation expenditures was based on the following assumptions:

- For equipment and buildings of *groups "A"* and *"B"*, depreciation deductions were determined by the same method as capital investments; and

- The useful life of the first wall is two years ($p_a = 0.5$); of the first blanket zone two years ($p_a = 0.1$); of other reactor construction elements 20 years ($p_a = 0.05$); and 30 years for buildings and facilities ($p_a = 0.033$).

The personnel expenditures were calculated by using the personnel coefficient $\varkappa = 1$ worker/MW with an average income level to be planned. Expenditures for routine maintenance amounted to 30 per cent of the depreciation deductions.

The calculations carried out for a TVE-2500 plant show that, compared with depreciation deductions and routine maintenance, fuel and operational costs on exploitation are really negligible, so that the annual expenses and finally the calculated costs turn out to be proportional to the size of specific capital investments.

One may then conclude that the calculated costs C for the TVE-2500, a fusion power station with a TOKAMAK reactor and a MHD system, amounts to 1.6 kopeck/kWh(e) given a level of capital investments of 470 Rub/kW(e).

Analoguous calculations were done for energy blocks with laser initiation of thermonuclear microexplosions (project LTB-500). The total specific capital investments for this plant amount to 370 Rub/kW(e), whereby in this case the laser system is the most capital-intensive element, its costs amount to roughly 40 per cent of the entire quantity of capital investments.

In calculating the annual expenses the coefficients of depreciation deductions were the following:

- Buildings $p_a = 0.033$
- Reactor $p_a = 0.33$
- Laser system $p_a = 0.13$
- Other equipment $p_a = 0.07$

The fuel component, which includes the costs of deuterium and lithium (without the first load), like in the foregoing case turned out to be a very small quantity (approximately 0.01 kopeck/kWh(e)).

The total quantity of calculated costs for laser fusion power plants LTR-500 at a level of capital investments of 370 Rub/kW(e) amounted to 1.4 kopeck/kWh(e).

It was shown in Appendix C-II that, in perspective, the maximum value of calculated costs for electric power plants operating on base load and located in the European part of the USSR, will be 1.2 to 1.4 kopeck/kWh(e).

The calculated costs for power plants of the types TVE-2500 and LTB-500 slightly exceed the maximum permissible values, but this amount is really not big. It should be mentioned that the results obtained are extremely sensitive to the cost of new equipment. This fact is illustrated in Figure C-1. It shows the dependence of the calculated costs on the specific cost of a superconducting magnetic system. With a specific cost of the superconducting magnetic system of approximately 35 to 40 Rub/kg the calculated costs will amount to 1.3 kopeck/kWh(e).

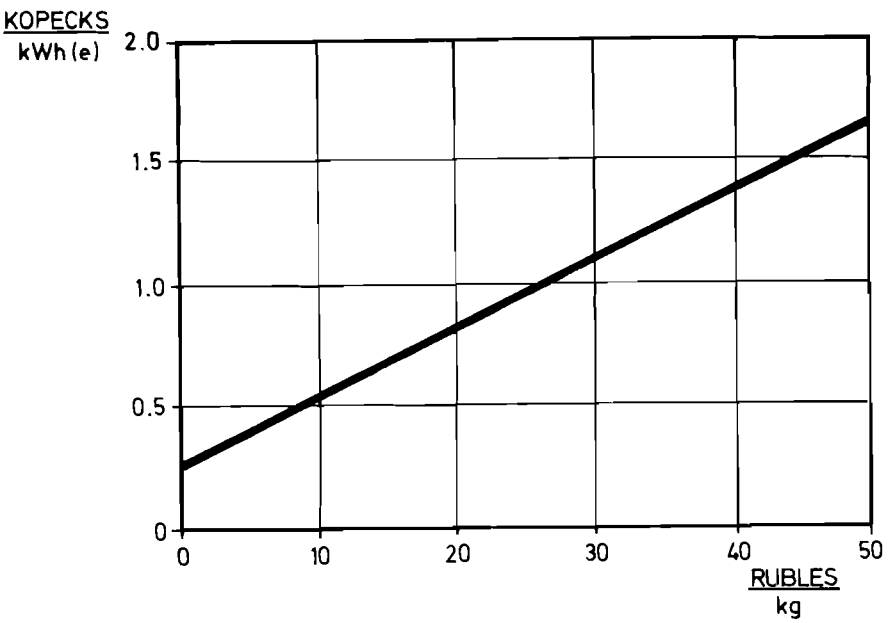

Figure C-1: Dependence of Quantity of Calculated Costs on Specific Cost of Superconducting Magnetic System (SCMS)

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APPENDIX D

LIST OF ABBREVIATIONS

ADAM + EVA	energy transmission system: $CH_4 + H_2O + 49$ kcal/mole $\rightleftarrows 3H_2 + CO$; EVA (German: *Einzelspaltrohr-Versuchs-Anlage*) is the chemical reactor for the production of $H_2 + CO$ (energy in)--its counterpart ADAM is the burner of the gaseous fuel (energy out)
AEC, USAEC	*United States Atomic Energy Commission*
AGR	*advanced gas-cooled* graphite-moderated *reactor*
AISI	*American Iron and Steel Institute*
ALCATOR	experimental TOKAMAK reactor (US)
ALCATOR-C	advanced version of ALCATOR (US)
ANL	*Argonne National Laboratory* (US)
ARGUS II ARGUS IV	US laser fusion test facilities
ASME	*American Society of Mechanical Engineers*
BELGONUCLEAIRE, BN	Belgian industrial consortium for the development of nuclear energy
BFS	a Soviet fast critical facility
BHP	*biological hazard potential*
BN 350	fast breeder prototype reactor (USSR)
BN 600	advanced fast breeder prototype reactor (USSR)
BNL	*Brookhaven National Laboratory* (US)

BOP	*balance of plant*
BOR 60	a Soviet experimental fast test reactor
BR	*breeding ratio*
BR-1	Soviet experimental fast reactors
BR-2	
BR-5	
BR-2 (Mol)	*Belgian Reactor Two*, research reactor, Mol (Belgium)
BWR	*boiling* light *water*-cooled and moderated *reactor*
CABRI	French for *he-goat*; pulsed experimental reactor (France)
CEA	*Commisseriat à l'Energie Atomique*, French atomic energy commission
CEGB	*Central Electricity Generating Board* (UK)
CFR-1	*Commerical Fast Reactor One* (UK)
10 CFR 20	typical citation from USAEC rules and regulations--*Code of Federal Regulations*, title 10, part 20
CLEMENTINE	the first Pu-fueled fast research reactor, (US)
CLEO	experimental TOKAMAK reactor (UK)
CPR	*commercial power reactor*
CRBR	*Clinch River breeder reactor*, US fast breeder demonstration (prototype) reactor
CTR	*controlled thermonuclear reactor*
CYCLOPS	experimental laser fusion test facility (US)
DBA	*design basis accident*
DEMO	*demonstration plant*
DFR	*Dounreay fast reactor*, fast breeder prototype reactor (UK)
DIDO	onomatopoeic for D_2O, research and test reactor using D_2O as moderator and coolant

DMFE	*Division of Magnetic Fusion Energy* (USERDA)
DOUBLET-II DOUBLET-III (D-III)	US experimental TOKAMAK reactors
dpa	*displacement per atom*
DPR	*demonstration power reactor*
EANDC	*European American Nuclear Data Committee*
EBR-I EBR-II	experimental breeder reactors (US)
EdF	*Electricité de France*, French national utility
EFFBR	*Enrico Fermi fast breeder reactor* (US)
ENDF	*evaluated nuclear data file* (for reactor application)
ENEL	*Ente Nazionale per l'Energia Elettrica*, Italian national utility
EPA	*Environmental Protection Agency* (US)
EPOXY	or epoxy resin, binding material, a flexible usually thermosetting resin made by polymerization of an epoxide
EPR	*experimental power reactor*
ERDA, USERDA	*United States Energy Research and Development Administration*
ETR	*engineering test reactor* (US)
EURATOM	*European Atomic Energy Community*
EVA + ADAM	see ADAM + EVA
FBR	*fast breeder reactor*
FBTR	*fast breeder test reactor* (India)
FCA	*fast critical assembly* (Japan)
FCEL	*fast critical experimental laboratory* (US)
FERF	*fusion engineering reactor facility* (US)

FFTF	*fast flux test facility*
FLIBE	*fluor-lithium-beryllium*, a liquid salt used in fusion reactors
FRG	*Federal Republic of Germany*
FR0	*Fast Reactor Zero* (Sweden)
FY	fiscal year (US)
GALAXY	code system used for producing microscopic group constants for reactor calculations (UK)
GCFR	*gas-cooled fast reactor*
GETR	*General Electric test reactor* (US)
GODIVA	after Lady Godiva, "naked" fast reactor (i.e. without reflector) excursion experiment
HAA-III	code used to calculate aerosol behavior in reactor containments (US)
Hc	critical magnetic field in a superconducting material
HEPA	*high-efficiency particulate air* (filter)
HFIR	*high-flux isotope reactor* (US)
HFNS	*high-flux neutron source* (US)
HFR	*high-flux reactor* (Netherlands)
HLW	*high-level waste*
HM	*heavy metal*
HTGR	*high-temperature gas-cooled* graphite-moderated *reactor*
IAEA	*International Atomic Energy Agency*
IBHP	*integrated biological hazard potential*
ICRP	*International Commission on Radiological Protection*
IIASA	*International Institute for Applied Systems Analysis* (Austria)

IMP	plasma research facility at the Oak Ridge National Laboratory (US)
INB	*Internationale Natrium Brutreaktor GmbH*, Belgian/Dutch/German industrial consortium founded for the construction of the SNR 300 fast breeder prototype reactor
INTERATOM	reactor manufacturer (FRG)
ISSEC	*internal spectral shifter and energy converter*
ITR	*ignition test reactor* (US)
JAERI	*Japan Atomic Energy Research Institute*
JANUS-I JANUS-II	laser fusion test facilities (US)
Jc	critical current in a superconducting material
JET	*joint European torus*
JOYO	fast experimental test reactor (Japan)
JT-60	experimental TOKAMAK test facility (Japan)
JTF-2	experimental TOKAMAK test facility (Japan)
KADIS	code used to calculate energy releases in hypothetical accidents of fast breeder reactors (FRG)
KEMA	*Keuring van Elektrotechnische Materialien*, Dutch testing institute
KNK I	sodium-cooled thermal test reactor (FRG)
KNK II	sodium-cooled fast test reactor with thermal driver zone (FRG)
LASL	*Los Alamos Scientific Laboratory* (US)
LD	*lethal dose*; e.g. $LD_{50/60}$: whole-body dose at which the probability of death is 50% within 60 days
LF	*load factor*
LFR	*laser fusion reactor*
LLL	*Lawrence Livermore Laboratory* (US)

LLW	*low-level waste*
LMFBR	*liquid-metal fast breeder reactor*
LWBR	*light-water breeder reactor*
LWR	*light water*-cooled and moderated *reactor*
MAGNOX	derived from magnesium alloy (e.g. Magnox 12); gas-cooled reactor using magnesium alloy as cladding material
MASURCA	*Maquette Surgeneratrice, Cadarache,* fast critical assembly (France)
MC^2	*multigroup cross-section constants,* code system used to produce microscopic group constants for reactor calculations (US)
MHD	*magneto-hydro-dynamic*
MIGROS	code system used for producing microscopic group constants for reactor calculations (FRG)
Mirror	type of magnetic confinement for high-temperature plasma in fusion reactors
MLW	*medium-level waste*
MONJU	fast breeder prototype reactor (Japan)
MPC	*maximum permissible concentration*
MSRE	*molten-salt reactor experiment* (US)
MTR	*materials test reactor* (US)
MX	*experimental Mirror* test facility (US)
Mylar	organic material that is sometimes used as thermal insulation material for low-temperature superconducting magnets
NAL	*National Accelerator Laboratory* (US)
NCRP	*National Council on Radiation Protection and Measurements* (US)
NERATOOM	Dutch industrial group for the development of nuclear power plants and plant components

NFS	*Nuclear Fuel Services*, a US reprocessing plant
NOL	*Naval Ordnance Laboratory* (US)
NPT	*Treaty on the Non-Proliferation of Nuclear Weapons*
OECD	*Organisation for Economic Co-operation and Development*
ORIGEN	*Oak Ridge isotope generation and depletion code*, code used to calculate the isotopic composition of irradiated reactor fuel (US)
ORMAK	experimental TOKAMAK test reactor (US)
ORNL	*Oak Ridge National Laboratory* (US)
PARDISEKO	code used to calculate aerosol behavior in reactor containments (FRG)
PDX	*poloidal divertor experiment*, experimental TOKAMAK test facility (US)
PE 16	nickel-base alloy
PEC	*Prova Elementi Combustibile*, fast experimental test reactor (Italy)
PEPR	*prototype experimental power reactor* (US)
PETULA	experimental TOKAMAK test facility (UK)
PF	*plant factor*
PFR	*prototype fast reactor* (UK)
PHENIX	fabulous bird connected with the worship of the sun, especially in ancient Egypt. As its end approaches it fashions a nest, and sets itself on fire. From this pyre miraculously springs a new phenix.
PLT	*Princeton large torus*, experimental TOKAMAK test facility (US)
PNC	*Power Reactor and Nuclear Fuel Development Corporation* (Japan)
PNURE	*preliminary national uranium resource evaluation* program (US)
PPPL	*Princeton Plasma Physics Laboratory* (US)

PULSATOR	experimental TOKAMAK test facility (FRG)
PWR	*pressurized* light-*water*-moderated and cooled *reactor*
QF	*quality factor*
RAPSODIE	experimental fast test reactor (France)
RBU	*Reaktor-Brennelement Union*, fuel-element manufacturer (FRG)
R&D	*research and development*
RF	*radio frequency*
RSS	*Reactor Safety Study* (Rasmussen report) (US)
RT	*room temperature*
RTNS	*rotating target neutron source*, neutron test facility (US)
RWE	*Rheinisch-Westfälische Elektrizitätswerk AG*, regional utility (FRG)
S/A	*subassembly*
SAEA	*Southwest Atomic Energy Associates*, US regional utility group
SAONE 1 and 2	commercial fast breeder power plants (France)
SAP	*sintered aluminium product safety reactor facility* (US)
SAREF	*safety reactor facility* (US)
SBK	*Schnellbrüter-Kernkraftwerk GmbH*, Belgian/Dutch/German utility group to operate SNR 300
S/C	*superconducting*
SCYLLA IV	experimental Theta Pinch test facility (US)
SEFOR	*Southwest experimental fast oxide reactor* (US)
SEP	Dutch utility group
SHIVA-I and II	proposed experimental laser test facilities (US)
SLSF	*sodium-loop safety facility* (US)

SNEAK	*Schnelle Nullenergie-Anordnung Karlsruhe*, fast critical zero power assembly (FRG)
SNR 300	*Schneller natriumgekühlter Reaktor*, fast breeder prototype reactor (FRG/Belgium/Netherlands)
SNR-2	fast breeder reactor power plant following upon the SNR 300 in the German/Belgian/Dutch breeder reactor program
SOFIRE	*sodium fire*, code used to calculate temperature and pressure build-up in case of sodium fires (US)
SRI	*Stanford Research Institute* (US)
SS	stainless steel (e.g. SS 304)
ST	early experimental TOKAMAK facility (US)
STEK	zero-power fast-thermal reactor (Netherlands)
SUPERPHENIX	fast breeder reactor power plant following upon PHENIX in the French breeder reactor program
SW	width over flats
SWU	*separative work unit*
SYNATOM	Belgian utility group
T-3,4,6,8,10,20	experimental TOKAMAK test facilities (USSR)
Tc	*critical temperature*
TETR	*TOKAMAK engineering test reactor* (US)
TFR	*toroidal fusion reactor* (US)
th.d.	*theoretical density*
Theta Pinch	type of magnetic confinement for fusion plasmas
TLV	*threshold limit value*
TM-3	experimental TOKAMAK test facility (USSR)
TNPG	*The Nuclear Power Group*, reactor manufacturer (UK)
TNS	*the next step*, proposed fusion reactor facility (US)

TOKAMAK	Russian: *toroidal chamber machine*, fusion reactor concept with a toroidal plasma configuration
TREAT	*transient reactor test facility* (US)
TTA	*toroidal test assembly*, superconducting-magnet test facility (US)
TVA	*Tennessee Valley Authority*, US regional utility
TZM	*titanium-zirconium-molybdenum*, a Mo-base alloy
UK	*United Kingdom*
UKAEA	*United Kingdom Atomic Energy Authority*
US (USA)	*United States of America*
USAEC	*United States Atomic Energy Commission*
USERDA	*United States Energy Research and Development Administration*
USSR	*Union of Soviet Socialist Republics*
UWMAK I,II,III	*University of Wisconsin* conceptual designs of commercial *TOKAMAK* reactors (US)
VENUS	code used to calculate energy releases in hypothetical accidents of fast breeder reactors (US)
VERA	*versatile experimental reactor assembly* (UK)
VIPER	*versatile intermediate pulsed experimental reactor* (UK)
2X 2XIIB	experimental test facilities based on the Mirror confinement scheme (US)
ZEBRA	*zero energy breeder reactor assembly* (UK)
ZPPR	*zero plutonium power reactor*, fast critical assembly (US)
ZPR-III,VI,IX	*zero power reactor III,VI,IX*, fast critical assemblies (US)

Units

appm	*atomic part per million*
atm	*atmosphere*; 1 atm $\hat{=}$ 1 kp/cm^2
bar	1 bar $\hat{=}$ 1.02 atm
barn	neutron cross-section unit; 1 barn $\hat{=}$ 10^{-24} cm^2
BTU	*British thermal unit*; 1 BTU $\hat{=}$ 0.252 kcal
c	*centi-*, $\hat{=}$ 10^{-2}
^{o}C	*degree centigrade*
cal	*calorie*
cap	*capita*
Ci	*curie*; 1 Ci $\hat{=}$ $3.7 \cdot 10^{10}$ decays/sec
cm	*centimeter*
cm^2	*square centimeter*
cm^3	*cubic centimeter*
day	*day*
$	*dollar*; in this paper dollar generally stands for US dollar
$	*dollar*: neutron reactivity value; 1$ is the difference in reactivity between delayed-neutron criticality and prompt-neutron criticality of a given reactor
(e)	*electric*
eV	*electronvolt*
g	*gram*
G	*giga-*, $\hat{=}$ 10^9
G	*gauss*, magnetic induction; 1 G $\hat{=}$ 10^{-4} V·sec/m^2
GJ	*gigajoule* = 10^9 J

GW	*gigawatt* = 10^9 W
h	*hour*
J	*joule*, 1J ≙ 1W·sec
k	*kilo-*, ≙ 10^3
K (^{0}K)	*kelvin*; absolute temperature scale in degrees centigrade
kcal	*kilocalorie*
keV	*kiloelectronvolt*
kg	*kilogram*
kJ	*kilojoule*
km	*kilometer*
km^3	*cubic kilometer*
kopeck	1/100 ruble (see ruble)
kp	*kilopond*
kW	*kilowatt*
kWh	*kilowatthour*
lb	*pound*, 1 lb ≙ 453 g
l	*liter*
m	*milli-*, ≙ 10^{-3}
m	*meter*
m^2	*square meter*
m^3	*cubic meter*
M	*mega-*, ≙ 10^6
Mach	ratio of a speed to the local sonic speed
MeV	*megaelectronvolt*

mill	*mill(s)*, 1 mill = $ 1/1000
min	*minute*
mm	*millimeter*
mol(e)	quantity of a substance whose weight is equal to the formula mass (in g)
mrem	*millirem* (see rem)
MW	*megawatt*
μ	*micro-*, ≙10^{-6}
μg	*microgram* = 10^{-6} g
μohm	*microohm* = 10^{-6} ohm
n	*neutron*
n	*nano*, ≙ 10^{-9}
nsec	*nanosecond* = 10^{-9} sec
ohm	*ohm*
p	*particle*
p	*pond*
%	*per cent*
ppm	*part per million*
rad	*radiation absorbed dose*
rem	*roentgen equivalent man*; 1 rem ≙ 1 rad·biological effectiveness; 1 rad ≙ 10^{-5} W·sec/g
rpm	*revolution per minute*
rub	*ruble* (USSR); being not freely convertible rubles are not converted to dollars; a very rough estimate would be: 1 rub ≙ $ 1.33
sec	*second*
t	*metric ton*, 1t ≙ 1000 kg
T	*tera-*, ≙ 10^{12}
T	*tesla*, magnetic flux density; 1T ≙ 1V·sec/m^2

(th)	*thermal*
TW	*terawatt* = 10^{12} W
V	*volt*
W	*watt*
yr	*year*