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Sodium Fast Reactor Fuels and Materials: Research Needs

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ABSTRACT

An expert panel was assembled to identify gaps in fuels and materials research prior to licensing sodium cooled fast reactor (SFR) design. The expert panel considered both metal and oxide fuels, various cladding and duct materials, structural materials, fuel performance codes, fabrication capability and records, and transient behavior of fuel types. A methodology was developed to rate the relative importance of phenomena and properties both as to importance to a regulatory body and the maturity of the technology base. The technology base for fuels and cladding was divided into three regimes: information of high maturity under conservative operating conditions, information of low maturity under more aggressive operating conditions, and future design expectations where meager data exist.

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This work was overseen and managed by Jeffrey L. LaChance (Sandia National Labs), who provided guidance on the approach taken, attended the expert elicitation panel meeting, and provided useful input during the report preparation. Recognition is also given to Tyrell Arment (MIT) for his assistance during the expert elicitation process.

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ACRONYMS

4S	Super-Safe, Small, Simple
ACRR	Annular Core Research Reactor
AFC	Advanced Fuel Campaign
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARC	Advanced Reactor Concepts
ASME	American Society of Mechanical Engineers
AT	Applied Technology
ATR	Advanced Test Reactor
CO_2	Carbon Dioxide
CRBR	Clinch River Breeder Reactor
DOE	Department of Energy
EBR-II	Experimental Breeder Reactor-II
FCCI	Fuel Cladding Chemical Interactions
FCMI	Fuel Cladding Mechanical Interactions
FFTF	Fast Flux Testing Facility
FM	Ferritic/Martensitic
HFIR	High Flux Isotope Reactor
IAEA	International Atomic Energy Agency
IFR	Integral Fast Reactor
IHX	Intermediate Heat Exchanger
INL	Idaho National Laboratory
JAEA	Japan Atomic Energy Agency
LMFBR	Liquid-Metal Fast-Breeder Reactor
LWR	Light Water Reactor
ORNL	Oak Ridge National Laboratory
OSTI	Office of Scientific and Technical Information
PIE	Post Irradiation Experiments
PNNL	Pacific Northwest National Laboratory
PRISM	Power Reactor Innovative Small Modular
R&D	Research and Development
SA	Severe Accident
S-CO_2	Supercritical-CO_2
SFR	Sodium Fast Reactor

SNL	Sandia National Laboratories
TREAT	Transient Reactor Test
U.S. NRC	U.S. Nuclear Regulatory Commission

I. OBJECTIVE

This document describes results of an expert opinion elicitation on research needed to ensure reliability and regulatory confidence of the potential fuels and structural materials to be used in the Sodium Fast Reactor (SFR). The expert opinion elicitation focused on a generic SFR design.

The purpose for the meeting of fast reactor fuels and materials specialists is to determine what R&D is required to license the two most mature fuels designs, metal and oxide fuel. Both fuel types have more than 50 years of experimentation and analyses the results of which are captured in voluminous publications and reports.

Expert opinions were elicited on the current state of knowledge for of the underlying phenomena affecting SFR fuels and structural materials performance. Experts were asked to rank these phenomena according to the:

- Importance of the phenomena with respect to regulatory and reliability concerns,
- State of experimental database, and
- State of current, quantitative understanding of the phenomena.

For this work, only nonproprietary, publically available data were used.

II. BACKGROUND

The two relatively mature fuel forms that have been considered for sodium reactor deployment are metal (U-Fs[*], U-Zr, U-Pu-Zr, U-TRU-Zr) and oxide ($U\text{-}O_2$, MOX, $U\text{-}O_2/TRU\text{-}O_2$). Metal fuels were the standard fuel for Experimental Breeder Reactor-II (EBR-II) for 30 years of operation, were used as a partial core loading in the Fast Flux Test Facility (FFTF) , and were the choice for the PRISM reactor where a licensing case was prepared. Several of the recent reactor designs have chosen metal fuel for their reference cores such as ARC-100, 4S, and TERRAPOWER. Oxide fuels were the standard fuel for FFTF and were the choice for the Clinch River Breeder Reactor (CRBR). In Japan, France, UK, Russia, and China, oxide fuels were the primary choice for their sodium-cooled fast reactors. Thus, a large knowledge base exists internationally for fast reactor oxide fuels.

The original vision for each fuel type was to start the first fast reactors with uranium-based fuel and then reprocess the fuel and blanket with a transition to a mixed oxide core (uranium/plutonium oxide) or a uranium/plutonium metal alloy core. Reprocessed fuel gave another dimension to the work needed to understand fuel performance due to the carryover from reprocessing of fission products and minor actinides from reprocessing.

All of the fuel research programs tended to follow similar paths. Small scale experiments, both in and out of the reactor, were conducted to arrive at the best design in terms of cladding and fuel combinations. During the course of these studies the most important phenomena were discovered. Fuel restructuring, fission gas release, the extent of fuel-cladding-chemical-interaction (FCCI) and fuel-cladding-mechanical-interaction (FCMI), and the extent of cladding swelling, creep, and embrittlement are examples of some of the important phenomena. In an effort to analyze, understand and predict these phenomena, property data were required for input into models and calculations.

With the availability of EBR-II and FFTF in the USA, and test reactors in other countries, a wide range of design variables and operation conditions were studied for full-size fuel assemblies. The individual fuel pin should be viewed as part of a fuel system where fuel pin bundle interaction, bundle duct interaction, and duct-duct interaction are as equally important to the performance as the performance individual pins.

With the availability of full-sized irradiated pins, the use of transient test reactors, such as TREAT in the USA, allowed the study of the behavior of irradiated fuel when subjected to relatively-severe over-power and loss-of-coolant-flow conditions. Transient testing of irradiated pins in hot-cell furnaces provided important complementary information.

Thus, the past fuels programs all contained the elements of in-core and ex-core experiments for both steady-state and transient conditions as well as the analyses and modeling of the important controlling phenomena.

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[*] Fissium (Fs) is nominally 2.4 wt% Mo, 1.9 wt% Ru, 0.3 wt% Rh, 0.2 wt% Pd, 0.1 wt% Zr and 0.01 wt% Nb, designed to mimic the noble metal fission products remaining after a simple reprocessing technique based on melt refinement

Over the past 50 years the progress in the development of fast reactor fuels has been continuous yet sporadic. During the 1960s and 1970s fast reactor fuel and reprocessing Research and Development (R&D) was intense worldwide with PUREX reprocessing and CRBR being the focus of U.S. programs that centered on oxide fuel. With the cessation of reprocessing, during the Carter administration, funding for these activities was greatly reduced. Fast reactor fuel development was revitalized in the 1980s with the introduction of the Integral Fast Reactor (IFR) program that included further metal fuel development and the pyro-reprocessing of the metal fuel. The IFR program was well-funded until 1994 when the Clinton administration curtailed fast reactor development with the closure of EBR-II and later FFTF.

During the latter part of the 1990s until today, interest reemerged in fast reactor fuel development not for breeding but with the realization that fast reactors are the best route to fissioning the minor actinides in commercial used fuel and thus to reduce the heat load and radio-toxicity of used fuel repositories. This new interest required understanding the performance of fast reactor metal and oxide fuels that contain a substantial concentration of minor actinides (Am, Np and Cm in particular). Much of the fuel development that is currently ongoing in the U.S. is involved with these issues.

When addressing issues associated with the gap analysis for the fuel, both oxide and metal, the knowledge gap becomes larger as the fuels are subjected to more demanding requirements. For example, UO_2 fuel or U-Zr fuel at up to 10 atom % (at%) burnup without plutonium, not reprocessed, no initial minor actinide addition, and at modest heat rating and temperature may require no additional R&D for licensing. However, more unresolved issues exist with reprocessed fuel that contains plutonium, minor actinides, and carryover fission products. Recent design requirements push the fuel systems to higher exposures than data exists for in the current suite of cladding and duct materials. These designs may reach regimes where excessive irradiation induced swelling, creep, and embrittlement become the controlling phenomena.

The resolutions of gaps which require the availability of a fast reactor test facility are difficult with no facilities in the U.S. and few in the world. The same is essentially true for safety related issues due to the unavailability of TREAT. The thermal spectrum Advanced Tests Reactor (ATR) reactor is somewhat suitable for special effects testing but falls short of irradiating full size fast reactor qualification assemblies.

Thus the subject gap analysis will be a graduated assessment that moves from the possible licensing of the basic oxide and metal fuel systems to the more complex systems where resolution of outstanding issues fall outside of the existing data base.

III. Expert Opinion Elicitation

In order to most efficiently direct future research efforts to create a licensing case for the SFR, an expert panel was assembled to identify gaps in the fuels and materials research areas which need to be filled before the Nuclear Regulatory Commission (NRC) can confidently license a sodium reactor. This panel's expertise covers both operational and experimental experience with fuel, fuel cladding, and structural materials. The panel briefly interacted via email prior to a 2.5-day meeting at Argonne National Laboratory to ensure that all relevant issues would be discussed in an orderly process. Figure 1 shows a high-level description of how the gap analysis was conducted. This approach was also used in the previous gap analysis reports (Corradini et al., 2010) (Sackett et al., 2010) (Powers et al., 2010) (Schmidt et al., 2011).

The degree of regulatory acceptability of the various fuel and materials issues was ranked qualitatively by the use of High (H), Medium (M) and Low (L) variables. High indicates that the regulatory body will require a high degree of confidence in the experimental database, materials knowledge or modeling techniques because the phenomena of interest can directly lead to a material failure. Medium indicates that the regulatory body will desire information about the phenomenon, but that the phenomenon is of secondary importance to understanding overall material performance and failure. Low indicates that understanding the phenomenon of interest is not instrumental to predicting material performance and thus only a basic understanding of the phenomenon is required for licensing.

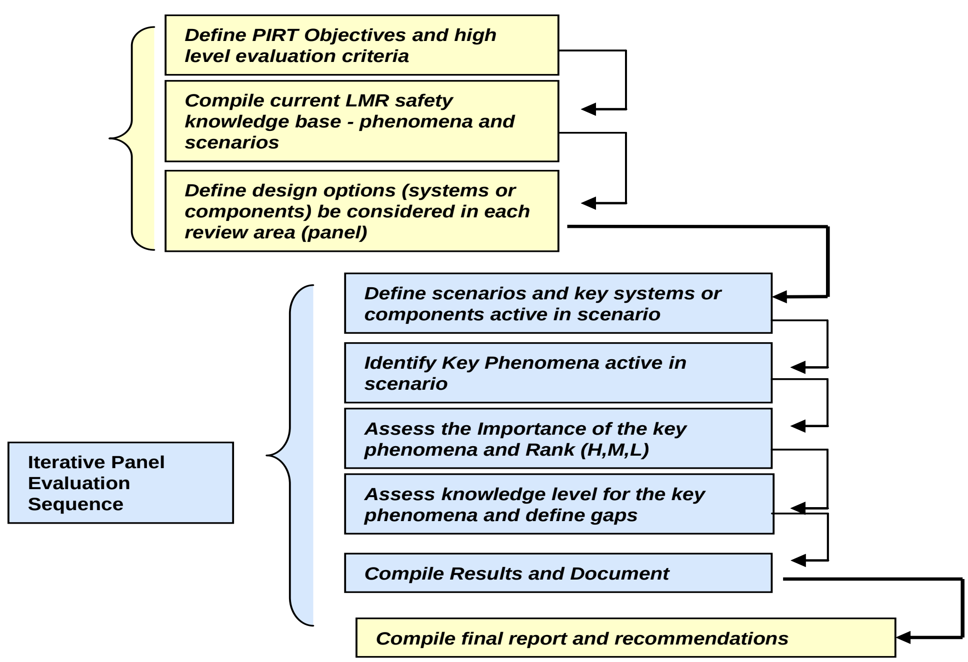

Figure 1. Sequence of gap analysis activities and panel process (Corradini et al., 2010).

Evaluating the state-of-knowledge of a phenomenon generally involves the assessment of both the modeling capabilities and the database to validate the model(s). The panel discussed each phenomenon extensively during the evaluation with the general criteria for state-of-knowledge for each level of the assessment defined as:

High (H)

- A physics-based or correlation-based model is available that adequately represents the phenomenon over the parameter space of interest.
- A database exists adequate to validate relevant models or to make an assessment.

Medium (M)

- A candidate model or correlation is available that addresses most of the phenomenon over a considerable portion of the parameter space.
- Data are available but are not necessarily complete or of high fidelity, allowing only moderately reliable assessments.

Low (L)

- No model exists, or model applicability is uncertain or speculative.
- No database exists; assessments cannot be made reliably.

Uncertain (U)

- Information available to the panel was inadequate to assess the state of knowledge.

The gap analysis knowledge results are also provided in the summary table, which includes comments for each ranking. In that same section of the report, we provide details of the rationale or justification for the panel knowledge ranking in our discussion.

IV. Evaluation of Gaps

The panel of experts opened the meeting with a general discussion of the state of knowledge for oxide and metal fast reactor fuels. Four fuel types were included in the discussion: fresh (which means not reprocessed) UO2, MOX, U-Zr, and U-Pu-Zr. The discussion was extended to include the additional issues associated with reprocessed fuel and fuel with additions of minor actinides.

The fuel performance issues that were addressed during the analysis included both steady-state and off-normal operation. Off-normal performance dealt with events that occur during steady-state operation such as load following and the behavior of fuel after cladding breach; i.e., run beyond cladding breach (RBCB) operation. In addition, the panel of experts was expanded to include experts knowledgeable in the area of accident-initiated events.

It was recognized that the performance characteristics of individual fuel pins which lead to cladding breach were not the only factor that determines the life of the reactor core. Fuel pin bundle interaction, which could restrict flow; bundle-duct interaction, which could affect fuel handling forces; and duct bow and dilation, which could affect both fuel handing forces and reactivity feedback were all concerns for core lifetime. However, the experts acknowledged that most of these concerns are design specific.

In the process of defining the important phenomena that impact fuel lifetime, the performance of cladding and duct materials may be controlling at high neutron irradiation exposures; i.e., high displacements per atom (dpa). Thus, in-depth discussion dealt with the state of knowledge for the current austenitic and Ferritic-martensitic cladding and duct materials as well as the use of advanced cladding and duct materials.

All the fast reactor test facilities in the US, which includes EBR-II, FFTF, and TREAT, have been shut down (TREAT, more specifically, being in “non-operational standby” status) for almost two decades. In the interim personnel have dispersed, and some records have been lost or stored in sub-standard conditions. This fact raised the question regarding how accessible and interpretable is the data base now and what condition has it been preserved for use in the future. This concern extended beyond irradiation data to the knowledge base for the procurement of materials and the fabrication of cladding, fuel, and duct components.

The expert panel first defined the Life-limiting phenomena for the fuel types at three burnup levels: 10 at%, 20 at%, and greater than 20at%. The rational for the three burnup levels was that for EBR-II (metal fuel) and FFTF(MOX fuel), the bulk of the irradiation data extended to about 10% because the exposure limits for the respective cores depended upon avoiding excessive fuel handling forces. For both oxide and metal fuel a limited number of assemblies were irradiated up to 20 at% burnup, but none beyond 20at%, even though some current reactor designs call for burnups greater than 20 at%. The state of knowledge was ranked according to the anticipated regulatory concern and then to the existing state of technical maturity and understanding. For metal fuel, virtually all the technical knowledge base belongs to the U.S.; whereas for oxide fuel, much of the knowledge base exists outside the US. It is uncertain how much of the foreign data would be available for U.S. reactor licensing.

Few if any knowledge gaps exist for fresh fuel of either oxide or metal fuel up to 10% burnup. However, the knowledge base is weak between 10 at% and 20 at% burnup and essentially non-existent beyond 20at% for fresh fuel. When reprocessed fuel was considered, the role of fission product carry-over, principally of lanthanide elements, resulted in identifying knowledge gaps. Further, if either oxide or metal fuel were to be utilized as hosts for minor actinides, then additional knowledge gaps were identified both with performance and fabrication.

Limiting phenomena were then identified for cladding and duct materials. The phenomena were ranked over four regimes: low dpa- low temperature, low dpa-high temperature, high dpa-low temperature, and high dpa-high temperature. High temperature was defined in this report to be up to than 630°C for cladding and 580°C for ducting. Low dpa was defined as less than 100 dpa and high dpa was defined as greater than 100dpa. Three materials were considered: 316 stainless steel, HT-9, and 9 Cr-1 Mo. By far the strongest data base for less than 100 dpa was 316, which is an adequate cladding and duct material for applications less than 100 dpa and nominal cladding temperatures less than 560°C. There were few knowledge gaps for 316 under these conditions. However, void swelling would limit its application to exposures less than 100 dpa. Modified forms of 316 such as alloy D9, designed to improve resistance to irradiation effects, can perform to ~100 dpa.

The Ferric-martensitic alloy HT-9 exhibits low swelling up to about 200 dpa, but there are no data beyond that neutron exposure and not a great deal up to 200 dpa. HT-9 may be a reasonable cladding and duct candidate for either fuel type up to reasonably high burnup (20 at%) but only at cladding temperatures below 600 °C due to the lower strength of HT-9 compared to 316 stainless steel. One potential gap identified for HT-9 is the lack of a vendor for the material. There appears to be an issue identifying a vendor who would be willing to become qualified to fabricate HT-9. Should a vendor be developed, then the question arises whether new heats of material would exhibit the same irradiation properties as the body of historical HT-9 irradiation data. This issue does not apply to type 316 stainless steel because this steel is a common fabrication material.

The martensitic material 9Cr-1Mo may solve both the issues of strength and swelling. However, less irradiation data exists for this class of alloys, and further, the qualification problem for potential vendors exists as is the case for HT-9.

After the Life-limiting phenomena were identified and ranked for fuel, cladding, and ducts, the important thermal and physical properties were identified and evaluated for gaps in knowledge by the same ranking system used for the phenomena. In general, the rankings mirrored the phenomena to which the property information applied. That is, for nominal fuel burnup and reasonably low dpa for the cladding and ducts, the property information was relatively well known. For higher exposures, gaps in the property information were evident.

The status of fuel modeling codes was discussed in an effort to identify gaps. The LIFE-metal code has been recently utilized routinely by a limited number of users. It was argued that the code does a reasonable job of describing existing irradiation data up to nominal burnups. The code is largely empirical and thus not useful for extrapolation to new operation regimes or new fuel designs. The main gap is that few people are knowledgeable enough to run the code and that substantial effort is required to document the code such that it can be transferred to new users.

The LIFE-oxide code (LIFE-4) is also being used routinely by a limited number of users and the documentation appears to be more complete than for LIFE-metal but some effort is required it to bring to current standards. For both codes documentation is required to describe the data base used to validate the codes.

A theme that ran through the entire meeting was whether or not the knowledge base for fast reactor fuels and materials was preserved intact. Fuel performance information is relatively available and retrievable through recent efforts to create computer searchable data bases. However, these data bases have stored publications and reports and not the original data. Whether or not the original data would be required in a licensing case was questionable. An attempt should be made to assess the availability and storage condition of original post irradiation data.

Operating information from EBR-II and FFTF is valuable to assess the performance of full assemblies. Duct bow and dilation measurements, assembly pull forces, and reactivity feedback information as a function of operating conditions are thought to exist at Pacific Northwest National Laboratories (PNNL) and Idaho National Laboratories (INL), but its location and condition need to be assessed.

Fabrication information for cladding, duct material, and fabrication information for both metal and oxide fuel exist in several locations. This information should be retrieved and assessed. Past practices would have to be duplicated to the extent possible for the existing data base to be valid for new fuels and materials.

Personnel capable of retrieving, assessing, and documenting outstanding information are ageing and leaving the workforce. Soon it will be nearly impossible to recognize and evaluate the value of existing data. Further, without the availability of testing facilities it will be impossible to duplicate subsets of the information that was generated over several decades and at great cost.

IV.A Presentation and Discussion of Rating Tables

The following tables were used as a means to first assess the importance of the various fuels performance characteristics to the regulatory licensing process and then assign a measure of the state of technologic maturity. For regulatory importance “H” indicated a characteristic of critical importance where the technologic maturity should also be “H” for successful licensing presentation. Where the regulatory importance was “H” and the technologic maturity was “L”, a definite knowledge gap exists. In the tables, the columns following the regulatory-concern column indicate the technology maturity levels.

The following categories were chosen for the tables as a means to envelop all possible licensing concerns.

1. Fresh metal and oxide fuel at 10 at%, 20 at%, and greater than 20 at% burnup.

2. Metal and oxide fuel with minor actinide additions at 10 at%, 20 at%, and greater than 20 at% burnup.
3. Metal and oxide fuel with carry-over of fission products from reprocessing at 10 at%, 20 at% and greater than 20 at% burnup.
4. Life-limiting phenomena and properties for 316 cladding.
5. Life-limiting phenomena and properties for HT-9 cladding.
6. Life-limiting phenomena and properties for advanced materials (e.g., 9Cr-1Mo or ferritic-martensitic steels).
7. Life-limiting phenomena and properties for 316 ducts.
8. Life-limiting phenomena and properties for HT-9 ducts.
9. Macroscopic thermal physical properties—metal UZr/UPuZr.
10. Macroscopic thermal physical properties—UO2/MOX

Table 1. Potential Life-Limiting Phenomena for Fresh Fuel

Fuel Phenomena	**Regulatory Concern, Metal/Oxide**	Metal, L.T. 10at%	Metal, L.T. 20at%	Metal, G.T. 20at%	Oxide, L.T. 10at%	Oxide, L.T. 20at%	Oxide, G.T. 20at%
Axial Growth	**L / (N/A)**	H	M	L	N/A	N/A	N/A
Fuel Swelling and FCMI	**H /M**	H	M	L	H	M	L
Gas Release	**H / H**	H	H	L	H	H	H
Fuel Constituent Redistribution	**M/ M**	H	M	L	H	M	L
FCCI	**H / M**	H	M	L	H	M	L
Fuel/Coolant Compatibility	**L / H**	H	H	H	H	L	L

Note: Experiment 496, a low smear density metal fuel test currently being irradiated, will increase our understanding of low smear density metal fuel

Table 2. Potential Life Limiting Phenomena for Fuel with Fission Product Carryover

Fuel Phenomena	**Regulatory Concern, Metal/Oxide**	Metal, L.T. 10at%	Metal, L.T. 20at%	Metal, G.T. 20at%	Oxide, L.T. 10at%	Oxide, L.T. 20at%	Oxide, G.T. 20at%
Axial Growth	**L / (N/A*)**	H	M	L	N/A	N/A	N/A
Fuel Swelling and FCMI	**H /M**	H	M	L	H	M	L
Gas Release	**H / H**	H	H	L	H	H	H
Fuel Constituent Redistribution	**M/ M**	H	M	L	H	M	L
FCCI	**H / M**	L	L	L	L	L	L
Fuel/Coolant Compatibility	**L / H**	H	H	H	H	L	L

*N/A – Not Applicable

Table 3. Potential Life Limiting Phenomena for Minor Actinide Bearing Fuel

Fuel Phenomena	**Regulatory Concern, Metal/Oxide**	Metal, L.T. 10at%	Metal, L.T. 20at%	Metal, G.T. 20at%	Oxide, L.T. 10at%	Oxide, L.T. 20at%	Oxide, G.T. 20at%
Axial Growth	**L / (N/A*)**	L	L	L	L	L	L
Fuel Swelling and FCMI	**H /M**	L	L	L	L	L	L
Gas Release	**H / H**	L	L	L	L	L	L
Fuel Constituent Redistribution	**M/ M**	L	L	L	L	L	L
FCCI	**H / M**	L	L	L	L	L	L
Fuel/Coolant Compatibility	**L / H**	L	L	L	L	L	L

*N/A – Not Applicable

Table 4. Phenomena and Properties for Stainless Steel 316 Cladding

Cladding Phenomena / Properties	**Regulatory Concern**	Low dpa (<100) / Low P.C.T.* (550-560°C)	Low dpa / High P.C.T. (~630°C)	High dpa (~200) / Low P.C.T.	High dpa / High P.C.T.
Creep Rate	**H**	H	H	IC***	IC
Swelling Rate	**H**	H	H	IC	IC
Fracture Toughness Properties	**L**	H	H	IC	IC
Yield Strength	**M**	H	H	IC	IC
Carbon Mass Transport	**M**	H	H	IC	IC
FCCI**	**M**	M	L	IC	IC

*P.C.T. – Peak Cladding Temperature, ** Only applicable to metal fuel, ,*** IC – Incompatible due to the poor high burnup performance of SS316 cladding.

Table 5. Phenomena and Properties for HT9 Cladding

Cladding Phenomena / Properties	**Regulatory Concern**	Low dpa (<100) / Low P.C.T.* (550-560°C)	Low dpa / High P.C.T. (~630°C)	High dpa (~200) / Low P.C.T.	High dpa / High P.C.T.
Creep Rate	**H**	H	M	H	L
Swelling Rate	**M**	H	M	H	L
Fracture Toughness Properties	**M**	H	M	H	L
Yield Strength	**M**	H	M	H	L
Carbon Mass Transport	**L**	N/A	N/A	N/A	N/A
FCCI**	**M**	H	M	H	M

*P.C.T. – Peak Cladding Temperature, ** Only applicable to metal fuel, ***N/A- Not Applicable
Note: Fabrication is not readily available, must be demonstrated to be consistent with historical HT9 database through mechanical and radiation testing.

Table 6. Phenomena and Properties for Advanced Cladding (e.g., 9Cr 1Mo, FMS)

Cladding Phenomena / Properties	**Regulatory Concern**	Low dpa (<100) / Low P.C.T.* (550-560^{o}C)	Low dpa / High P.C.T. (~630^{o}C)	High dpa (~200) / Low P.C.T.	High dpa / High P.C.T.
Creep Rate	**H**	M	M	M	L
Swelling Rate	**M**	M	M	M	L
Fracture Toughness Properties	**M**	M	M	M	L
Yield Strength	**M**	M	M	M	L
Carbon Mass Transport	**L**	N/A***	N/A	N/A	N/A
FCCI**	**M**	L	L	L	L

*P.C.T. – Peak Cladding Temperature, ** Only applicable to metal fuel, ***N/A- Not Applicable
Note: Fabrication is difficult but organizations claim that they can fabricate on an industrial scale.
Note: Japan and France have data from the phenomena/properties listed above, but it is unclear how available this data would be to a U.S. designer.

Table 7. Phenomena and Properties for Stainless Steel 316 Duct

Duct Phenomena / Properties f(dpa,T)	**Regulatory Significance**	Low dpa (<100) / Duct Inlet Temperature (400°C)	Low dpa (<100) / Duct Outlet Temperature (550°C)	Low dpa (<100) / Peak Duct Temperature (~580°C)	High dpa (~200) / Duct Inlet Temperature (400°C)	High dpa (~200) / Duct Outlet Temperature (550°C)	High dpa (~200) / Peak Duct Temperature (580°C)
Creep Rate	**M**	H	H	H	IC*	IC	IC
Swelling Rate	**M**	H	H	H	IC	IC	IC
Fracture Toughness Properties	**L**	H	H	H	IC	IC	IC
Yield Strength	**L**	H	H	H	IC	IC	IC
Carbon Mass Transport	**L**	H	H	H	IC	IC	IC
Dimensional Distortion	**H**	H	H	H	IC	IC	IC
Bundle Interaction	**H**	M**	M**	M**	IC	IC	IC
Bundle-Duct Interaction	**H**	M**	M**	M**	IC	IC	IC
Duct-Duct Interaction	**M**	H**	H**	H**	IC	IC	IC

* IC – Incompatible, ** If information has been preserved

Table 8. Phenomena and Properties for HT9 Duct

Duct Phenomena / Properties f(dpa,T)	**Regulatory Significance**	Low dpa (<100) / Duct Inlet Temperature (400°C)	Low dpa (<100) / Duct Outlet Temperature (550°C)	Low dpa (<100) / Peak Duct Temperature (~580°C)	High dpa (~200) / Duct Inlet Temperature (400°C)	High dpa (~200) / Duct Outlet Temperature (550°C)	High dpa (~200) / Peak Duct Temperature (580°C)
Creep Rate	**M**	H	H	H	M	M	M
Swelling Rate	**M**	H	H	H	M	M	M
Fracture Toughness Properties	**H**	H	H	H	M	M	M
Yield Strength	**L**	H	H	H	M	M	M
Carbon Mass Transport	**L**	H	H	H	H	H	H
Dimensional Distortion	**H**	H	H	H	M	M	M
Bundle Interaction	**H**	M*	M*	M*	M*	M*	M*
Bundle-Duct Interaction	**H**	M*	M*	M*	M*	M*	M*
Duct-Duct Interaction	**M**	H*	H*	H*	M*	M*	M*

* If information has been preserved

Table 9. Macroscopic Metal Fuel Thermal Physical Properties (UZr / UPuZr)

Physical Properties	**Regulatory Significance**	Low BU (<10%)	High BU (>10%)
Thermal Conductivity	**H**	H / H	L / L
Heat Capacity	**H**	H / H	L / L
Cladding Comp. Diffusivity	**M**	H / H	L / L
Free Energy of Formation	**L**	H / H	H / H
Phase Relationships	**H**	H / M	H / M
Primary Comp Diffusivity	**M**	M / M	L / L
Minor Actinide Diffusivity	**M**	L / L	L / L
Yield Strength	**L**	L / L	L / L
Thermal Creep Rate	**L**	L / L	L / L
Radiation Creep Rate	**L**	L / L	L / L
Young's Modulus	**L**	M / L	L / L
Thermal Expansion	**H**	H / H	L / L
Poisson's Ratio	**L**	M / M	L / L
Hardness	**L**	M / M	L / L

Table 10. Macroscopic Oxide Fuel Thermal Physical Properties (UO_2 / MOX)

Physical Properties	**Regulatory Significance**	Low BU (<10%)	High BU (>10%)
Thermal Conductivity	**H**	H / H	M / M
Heat Capacity	**H**	H / H	L / L
Cladding Comp. Diffusivity	**M**	IC / IC	L / L
Free Energy of Formation	**H**	H / H	L / L
Phase Relationships	**M**	H / H	L / L
Primary Component Diffusivity	**M**	L / L	L / L
Minor Actinide Diffusivity*	**L**	L / L	L / L
Yield Strength	**L**	M / L	L / L
Thermal Creep Rate	**L**	H / M	L / L
Radiation Creep Rate	**L**	H / M	L / L
Young's Modulus	**H**	H / M	L / L
Thermal Expansion	**L**	H / H	L / L
Poisson's Ratio	**L**	H / M	L / L
Hardness	**H**	H / M	L / L

*Note: Much of the high burnup data is Japanese or French in origin

Due to the unavailability of the proper expertise during the panel meeting, regulatory gaps in structural materials were not directly considered by the panel. Instead, this report leveraged a number of previous studies which considered the current state of SFR structural materials. Technology status evaluations for materials in various components and environments can be found in Tables 11- 15 (Chopra and Natesan, 2007).

Table 11. Reactor System Structural Components: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
A-1	Vessel	Stainless steel (Type 316)	Primary sodium and argon gas or air	Thermal aging Irradiation Weld integrity	Neutron fluence Service temperature Service life	Good	Sufficient	1
A-2	Vessel Enclosure	Stainless steel (Type 316)	Primary sodium & argon?	Thermal aging Irradiation Weld integrity	Neutron fluence Service temperature Service life	Good	Sufficient	1
A-3	Rotatable Plug for Reactor Head	?	?	?	?	?	?	?
A-4	Guard Vessel	Stainless steel (Type 316), Fe-9Cr-Mo Steel	Argon gas and leaking sodium	Corrosion	Temperature Exposure time Oxygen content in sodium	Good	Sufficient	1
A-5	Core Support Structure	Stainless steel	Primary sodium	Irradiation Thermal aging Crevice corrosion	Temperature Exposure time Crevice chemistry Oxygen content in sodium	Good	Sufficient	1

Table 12. Primary Heat Transport System: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
B-1	Internal Piping	Stainless steel (Type 316)	Primary sodium	Corrosion Carburization Radioactive mass transport	Temperature Delta T Oxygen and carbon in sodium Sodium flow velocity Sodium purification capability	Good	Fairly good Needs system assessment for ABR regarding dynamic carbon level	2
B-2	Mechanical Pump (Impeller, Diffuser)	Stainless steel (Type 316)	Primary (388°C) or Secondary (344°C) Sodium	Corrosion Fatigue resistance	Flow velocity Vibration Applied load Sodium purity Temperature	Good, need to identify vendors	Fairly good	2
B-3	Electromagnetic Pump	TBD	Primary (388°C) or Secondary (344°C) Sodium	Corrosion Fatigue resistance Electrical compatibility	Flow velocity Vibration Applied load Sodium purity Temperature Electrical interference	Unknown	Poor	3
B-4	Intermediate Heat Exchanger Shell	SS (Type 304 or 316)	Primary Sodium inside (510/355°C)	Sodium Corrosion Swelling Thermal Creep Irradiation Creep Fatigue and creep-fatigue Interstitial Element transfer	Oxygen and carbon in sodium Service life Temperature Mechanical load	Good	Adequate	2
B-5	Intermediate Heat Exchanger Tubes	SS (Type 304H), Fe-9Cr-Mo Steel	Secondary Sodium inside (333/488°C)& Primary Sodium outside (510/355°C)	Sodium Corrosion Swelling Thermal Creep Irradiation Creep Fatigue and creep-fatigue Interstitial Element transfer	Oxygen and carbon in sodium Service life Temperature Mechanical load	Good	Adequate	2

Table 13. Secondary Heat Transport System: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
C-1	Secondary System Pump	TBD	Secondary sodium	Corrosion Fatigue resistance	Flow velocity Vibration Applied load Sodium purity Temperature	Good, need to identify vendors	Fairly good	2
C-2	Sodium Piping	SS (Type 316)	Secondary sodium	Corrosion Carburization Radioactive mass transport	Temperature Delta T Oxygen and carbon in sodium Sodium flow velocity Sodium purification capability	Good	Adequate	2

Table 14. Power Conversion System, Supercritical CO2 Brayton Cycle: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
D-1.1	Compressor	TBD	CO_2, moisture, impurities?, high pressure	Oxidation Carburization Creep Fatigue Creep fatigue	Temperature Gas purity Applied load	Probably adequate	Limited	3
D-1.2	Turbine Generator	TBD	CO_2, moisture, impurities?, high pressure	Oxidation Carburization Creep Fatigue Creep fatigue	Temperature Gas purity Applied load Gas velocity	Unknown	Seriously lacking	3
D-1.3	Sodium to CO_2 Heat Exchanger	TBD	CO_2, moisture, high pressure	Oxidation Carburization Creep Fatigue Creep fatigue Sodium-CO_2 reaction	Temperature Delta T Gas purity Applied load Tube/channel failure Plugging Thin section material	Unknown	Seriously lacking	3
D-1.4	Recuperator	TBD	CO_2, moisture	Oxidation Carburization Creep Fatigue Creep fatigue	Temperature Gas purity Applied load	Adequate?	Probably good	2

Table 15. Power Conversion System, Steam Rankine Cycle: Technology Status 1:Adequate; 2:Needs more work, 3:Almost no data (Chopra and Natesan, 2007)

Item	Structure & Component	Material	Environment	Degradation Process or Mechanism	Factors Controlling Occurrence	Fabrication Capability	Knowledge/ Database	Technology Status
D-2.1	Steam Generator Shell	Ferritic Steel (Fe-2 1/4Cr-1Mo)	Secondary Sodium (502-344°C)	Sodium corrosion Interstitial element transfer Thermal aging Sodium-water reaction Caustic effect Thermal creep Fatigue & creep-fatigue	Sodium purity Temperature Delta T Steam leak Applied load Transients Long term aging	Good	Fairly good	2
D-2.2	Steam Generator Tubing	Ferritic Steel (Fe-2 1/4Cr-1Mo)	Water or Steam inside (287-482°C at ≈11 MPa) & Secondary Sodium outside (502-344°C)	Sodium Corrosion Interstitial Element transfer Thermal aging Sodium-water reaction Caustic effect Thermal creep Fatigue & creep-fatigue	Sodium purity Temperature Delta T Steam leak Applied load Transients Long term aging	Good	Fairly good	2
D-2.3	Hot Leg Steam Piping	Ferritic Steel (Fe-2 1/4Cr-1Mo)	Steam (482°C)	Flow-Assisted Corrosion Fatigue & creep-fatigue Thermal aging	Flow velocity Steam pressure Temperature Service time Thermal aging Water chemistry	Good	Fairly good	2
D-2.4	Cold Leg Steam Piping	Carbon Steel (SA 106 Gr. B)	Treated Water (287°C)	Flow-Assisted Corrosion General corrosion Fatigue	Flow velocity Steam pressure Temperature Service time Thermal aging Water chemistry	Good	Fairly good	1
D-2.5	Steam Turbine	Ferritic Steel (Intermediate chromium)	Steam	Steam oxidation Scale exfoliation Creep and fatigue Flow induced corrosion	Temperature Service time Applied load Steam flow velocity	Good	Fairly good	1

IV.B Discussion of Metal Fuels

Uranium-based metallic fuels have been used as the driver fuel for multiple SFRs, including the Experimental Breeder Reactor-I, the Dounreay Fast Reactor, the Enrico Fermi Fast Breeder Reactor, and most recently over 30 years of operation in the Experimental Breeder Reactor-II (EBR-II). Furthermore, the FFTF performed an extensive set of irradiations of metallic fuel qualification subassemblies and was poised to convert its core to a metallic driver fuel just prior to its shutdown. In EBR-II, U-10Zr metallic driver fuels operated reliably to 10 at% burnup, with extensive experimental testing of U-Zr and U-Pu-Zr metallic fuels to burnups of 20 at% conducted in both EBR-II and FFTF. It is not surprising, therefore, that the conviction of the fuels experts participating in this Gap Analysis was that the data base for a licensing case for metallic fuels (especially U-10Zr, but extending in large measure to U-20Pu-10Zr) is, in general, strong for burnups up to 10 at%, decreasing above that burnup level due to the reduced amount of experimental data.

IV.B.1 Life-limiting Phenomena

The major irradiation performance phenomena having the potential to limit the life, or the reliable performance, of metallic fuels are:

- Axial Growth
- Fuel Swelling & FCMI
- Gas Release
- Fuel Constituent Redistribution
- FCCI
- Fuel-coolant Compatibility

These irradiation performance phenomena were shown in Table 1 along with the consensus of the fuels experts as to their technological maturity level assessed from the perspective of regulatory importance. The rationale behind each score is briefly described in the subsequent paragraphs.

<u>Axial Growth</u>. Axial growth is of low concern from a regulatory perspective. U-Zr fuels can grow considerably during the first few at% burnup (i.e., as much as 10 at%), although U-Pu-Zr fuels exhibit considerably less growth. This phenomenon is more of an operational concern rather than a safety concern. Axial growth of metallic fuel is a source of negative reactivity for the core, for which reactor operations must be able to compensate. Extensive data have been collected on axial growth as a function of burnup in metallic fuels and reported in the literature (Hofman and Walters, 1994).

<u>Fuel Swelling & FCMI</u>. Metallic fuel is well known to be a high swelling fuel form under SFR conditions, with essentially no difference between U-Zr and U-Pu-Zr alloys. In the early days of metallic fuels, which were fabricated with either no fuel-cladding gap or a very small gap, fuel swelling quickly led to extensive FCMI resulting in fuel failure at low burnups (< 2 at%). However, it was eventually learned that this high swelling behavior is driven by rapid fission gas bubble nucleation and growth, which if allowed to swell unconstrained will result in an interconnection of bubbles and release of a large fraction of the fission gases produced in the fuel

(~80%). This interconnection phenomenon is strictly a geometrical effect that occurs at 33 vol.-% swelling, which for metallic fuels is reached at 2-3 at% burnup, after which continued accumulation of solid fission products drives further fuel swelling at a greatly reduced rate. Modern metallic fuel designs make use of a large fuel-cladding gap (i.e., 75% smear density) which allows this point of dramatic reduction in fuel swelling to occur prior to FCMI; with this design accommodation, FCMI typically does not threaten to limit metallic fuel lifetime until well over 10 at% burnup. Extensive data has been collected on metallic fuel swelling as a function of burnup in metallic fuels and has been reported in the literature (Hofman and Walters, 1994).

Gas Release. As discussed under the Fuel Swelling & FCMI heading, modern metallic fuel designs allow for early interconnection of fission gas bubbles, resulting in fission gas release values of approximately 80% above a few percent burnup. This leads to a need for a large fission gas plenum to accommodate such high fission gas release. Too small of a plenum can result in creep rupture of the cladding being the most significant Life-limiting irradiation performance phenomenon in metallic fuels, while too large of a plenum can be a significant economic penalty; this is a design trade off issue that can raise regulatory concerns. Nevertheless, extensive data has been collected on metallic fuel gas release as a function of burnup in metallic fuels and has been reported in the literature (Hofman and Walters, 1994).

Fuel Constituent Redistribution. Both U-10Zr and U-20Pu-10Zr metallic fuel systems undergo fuel constituent redistribution in the radial fuel dimension due to the fact that the traditional fuel temperature operating regimes span a miscibility gap in both alloy systems. The behavior is not identical in the binary and ternary systems, although both can result in radial zones having depleted Zr content under irradiation. Since the Zr content of the metallic fuel alloys is largely responsible for keeping the fuel solidus temperature high, this raises the regulatory concern that the solidus temperature is reduced in any Zr-depleted radial zone. The safety case for a metallic fuel must take this into account in demonstrating thermal margin under all operational scenarios. Data on constituent redistribution of irradiated metallic fuels is difficult to obtain, so less has been collected and reported in the literature than for most of the other phenomena discussed. Nevertheless, the theoretical understanding of the phenomenon seems to be well known, and several models have been developed that appear to adequately explain the reported experimental data (Hofman et al., 1996) (Kim et al., 2006).

Fuel-cladding Chemical Interaction. FCCI in metallic fuels results primarily from lanthanide fission products (i.e., La, Ce, Pr, Nd, Sm) that transport through the fuel and react with stainless steel cladding alloys. There is an incubation period associated with birth and transport of the fission products, after which the cladding reaction seems to follow a typical Arrhenius dependence on temperature. The reaction that occurs on the cladding inner surface produces a brittle interaction layer that grows with burnup and is generally considered as wastage. Thus, FCCI acts to thin the cladding wall, thus increasing the cladding stress, which must be accounted for in cladding creep rupture assessments. Nevertheless, it has not served to limit metallic fuel lifetimes for burnups to 10 at% and peak cladding temperatures less than 600°C. Extensive data has been collected on metallic fuel-stainless steel cladding chemical interaction as a function of burnup and has been reported in the literature (Keiser, 2009).

Fuel-coolant Compatibility. Fuel-coolant compatibility is a non-issue for metallic fuels in sodium-cooled reactors. Metallic fuels generally include liquid sodium as a thermal bonding agent in the fuel-cladding gap, and extensive run-beyond-cladding-breach testing for metallic fuels was performed in EBR-II. Metallic fuel is totally compatible with sodium (Crawford et al. 2007).

IV.B.2Thermo-physical Properties

Since the licensing case for a nuclear fuel is made with considerable reliance on analysis and modeling of fuel behavior under reactor conditions, those thermo-physical properties needed for such analyses are very important. Typically, either experimental measurements or conservative assessments are required to support a safety or licensing case for any nuclear fuel. Table 9 shows the thermo-physical properties of importance in metallic fuel analyses, although they are not all of equal importance. The most critical are those properties that support the thermal analysis of fuel under irradiation and assessments of limiting conditions. For metallic fuels, these are: thermal conductivity, heat capacity, thermal expansion, and phase relationships. Thermo-physical properties beyond these are either of minimal regulatory significance or are easily estimated with adequate conservatism. In general, adequate knowledge of all the thermo-physical properties important to developing a licensing case for metallic fuels seems to be in hand for low (< 10 at%) burnups.

Thermal Conductivity. Knowledge of the thermal conductivity is vital for any nuclear fuel since calculated fuel temperatures are directly proportional to it. For metallic fuels, thermal conductivity is a function of alloy content, temperature, and burnup. Thermal conductivity for both U-Zr and U-Pu-Zr metallic fuels have been widely determined experimentally as a function of temperature and reported in the literature. Experimental determination of thermal conductivity for irradiated metallic fuels, though, has apparently been estimated in only one set of measurements (Bauer and Holland, 1995). Methods for determination of the thermal conductivity with burnup, which would appear to be conservative, have been reported. In any event, the relatively low temperature at which metallic fuels typically operate, with considerable thermal margin to the solidus temperature, means considerable uncertainty on the effect of burnup should be able to be accommodated.

Heat Capacity. Knowledge of the heat capacity of a nuclear fuel is needed for transient thermal analyses. Heat capacity for both U-Zr and U-Pu-Zr metallic fuels has been widely determined experimentally as a function of temperature and reported in the literature. Analyses have generally assumed that heat capacity does not change with burnup.

Thermal Expansion. Knowledge of the thermal expansion of a nuclear fuel is needed for both steady-state and transient thermal and thermo-mechanical analyses. Thermal expansion for both U-Zr and U-Pu-Zr metallic fuels have been widely determined experimentally as a function of temperature and reported in the literature. Analyses have generally assumed that thermal expansion does not change with burnup.

Phase Relationships. Knowledge of the phase relationships of a metallic nuclear fuel is needed for both steady-state and transient thermal analyses. Specifically, the solidus temperature as a function of alloy composition is taken as the effective melting temperature of a metallic fuel, and therefore represents a limiting condition from a regulatory perspective. Phase diagrams for both

U-Zr and U-Pu-Zr metallic fuels have been constructed with experimental validation and reported in the literature. Analyses have generally, but not always, assumed that the solidus temperature does not change with burnup.

IV.B.3 Notes on Metallic Fuels with Minor Actinide Additions

Interest in metallic fuels for actinide burning applications, for which Np and Am are incorporated up to a few percent into the fuel at fabrication, began in the early 1990's right at the time EBR-II and FFTF operations were terminated. Thus, there has not been extensive testing of metallic fuels with minor actinide additions. One experiment was performed in EBR-II prior to its shutdown that incorporated Np and Am into U-Pu-Zr metallic fuel (i.e., X501). It was irradiated to 8 at% burnup without failure. Post-irradiation examination revealed that considerable He gas was generated by the transmutation of Am and resulting decay chains, which was released at 90%. Radial redistribution of the Am, similar though not identical to Zr, was also observed. Fuel-cladding chemical interaction was apparently not affected in a measureable way by the presence of the minor actinides (Meyer et al., 2009). Nevertheless, gas release (i.e., He) will be increased and constituent redistribution will be affected by the addition of minor actinides to metallic fuels. While additional testing of metallic fuels with minor actinide additions is on-going in the Advanced Test Reactor, these tests are not entirely prototypic of a fast reactor environment and have not yet been fully assessed (MacLean and Hayes, 2007). Thus, it is acknowledged that the data is likely not currently in hand to license a metallic fuel with minor actinide additions (see Table 3).

IV.B.4 Notes on Metallic Fuels with Fission Product Carry-over

The primary concern with the licensing case for metallic fuels resulting from recycle using an electro-chemical process is the anticipation of carry-over of some lanthanide fission products, perhaps as much as 1 wt % in re-fabricated metallic fuels. As these elements are those primarily responsible for FCCI, the obvious concern is the FCCI could be accelerated for metallic fuels fabricated from recycle feed streams. This is an area of current research activities (Mariani et al., *in press*). Thus, it is acknowledged that the data is likely not currently in hand to license a metallic fuel fabricated using recycle feed streams (see Table 2).

IV.C Discussion of Oxide Fuels

As well as fueling all light-water reactors (LWRs) worldwide, uranium-based oxides have been used extensively as the driver fuel for several sodium fast reactors (SFRs) in Russia and Kazakhstan, including the BOR-10, BOR-60, BN-350, and BN-600 reactors; the latter 600-MWe SFR, for example, is currently in its thirty-second year of full power operation with stainless steel-clad UO_2 pellet fuel and with an enviable plant factor.

Plutonium-bearing mixed-oxide (MOX) fuel has been used as driver fuel in a wider range of SFRs: in the Southwest Experimental Fast Oxide Reactor (SEFOR) and the Fast Flux Test Facility (FFTF) in the U.S.; in the U.K. Prototype Fast Reactor (PFR); in the Rapsodie, Phenix and Superphenix reactors in France; in the German KNK-II reactor; and in the JOYO and MONJU reactors in Japan.

Additionally, extensive domestic fuels irradiation programs were performed in the Experimental Breeder Reactor-II (EBR-II) and the FFTF to determine Life-limiting phenomena in MOX fuel, including (in EBR-II) mild transient behavior of the fuel, and its potential for operating with breached cladding. Reviews of this U.S. work were given by Lambert and Strain (1994) and by Crawford et al (2007). It is no wonder, given this significant domestic and foreign experience, that fuels experts participating in this Gap Analysis were convinced that the case for licensing SFR oxide fuel is strong. Also, the domestic experience regarding the non-Pu-bearing material UO_2 is substantially less than with its Pu-bearing counterpart $(U,Pu)O_2$; however, the broad database and understanding developed for LWRs with UO_2 fuel can be extrapolated to SFR conditions without difficulty.

IV.C.1 Life-limiting Phenomena

Major phenomena that may have potential to limit reliable performance of oxide fuels are:

- Axial growth
- Fuel swelling and fuel-cladding mechanical interaction
- Fission gas release
- Fuel constituent redistribution
- Fuel-cladding chemical interaction
- Fuel-coolant compatibility

These irradiation performance phenomena are shown in Table 1 along with the consensus of the fuels experts as to their technological maturity level assessed from the regulatory viewpoint. The rationale for each score is briefly described.

Axial Growth: Axial growth nowadays is of low concern from a regulatory perspective. This was not the case in the early days of SFR development in the U.S. In fact, the SEFOR reactor was built specifically to check reactivity feedback in an oxide core (particularly the Doppler effect) and to determine the extent of axial growth in this comparatively new fuel type (Noble et al., 1972). In contrast to metallic fuels with substantial axial growth (≥10 at%), oxide fuels exhibited less than 1% change in overall length to a significant burnup. For this reason, axial growth is considered to be a non-applicable (N/A) phenomenon for oxide fuels.

Fuel Swelling and Fuel-Cladding Mechanical Interaction (FCMI): Oxide fuels are well known as medium swelling fuel forms, UO_2 and MOX exhibiting similar behavior. At temperature below about 1000^oC, the swelling of both fuels is in the region of 1.7% $\Delta V/V$ per at% burnup. At higher temperatures, where fission gas release occurs due to thermally-induced equiaxed and columnar grain growth, a value nearer 1.0% $\Delta V/V$ per at% burnup applies. Such fuel swelling can be partly accommodated by porosity in the sintered fuel and by additional porosity associated with unhealed thermal cracks in the fuel pellets and the residual fuel-cladding gap.

With judicious choice of fuel density and fuel-cladding gap size, the incidence of FCMI can be minimized in SFR oxide fuel elements to high burnup. For this reason fuel swelling and FCMI are deemed Life-limiting phenomena of medium regulatory concern. The experience level with these phenomena is considered to be high for burnups below 10 at%, medium for burnups between 10 and 20 at%, and low for burnups above 20 at% [being limited to data on a small number of examined fuel elements from the Core Demonstration Experiment (CDE) in the FFTF in which peak burnups of 24 at% were achieved (Bridges et al., 1993)].

Gas Release: As ceramics with low thermal conductivity, SFR oxide fuels operate at high temperatures and release significant quantities of the fission gas generated in them. At linear powers of 25-30 kW/m (and higher), fission gas release exceeds 50% at low burnup ($\leq$5 at%), and increases steadily with increasing burnup as linkage of grain-boundary bubbles occurs in the fuel interior. Fission gas release is then typically 75% and higher.

This phenomenon is considered of high priority in licensing for the simple reason that if insufficient plenum volume is available stresses in the cladding can be high enough to cause significant creep and possibly lead to pin failure. The confidence level in the data on this phenomenon is considered high at all burnup levels. For example, at burnups above 20 at% where data are limited, a conservative value of 100% gas release can be assumed for licensing purposes.

Fuel Constituent Redistribution: UO_2 and $(U,Pu)O_2$ are single phase to their melting point, so that phase changes *per se* do not lead to constituent redistribution, which they do in metal fuels. However, redistribution does take place by vapor transport and is strongly affected by fuel stoichiometry. In hypostoichiometric MOX, i.e., $(U,Pu)O_{2-x}$ normally used in SFRs, Pu species tend to move up the temperature gradient and U species down the temperature gradient.

Such redistribution will result in mild radial changes in the heat production rate in-reactor. In turn, this can increase fuel centerline temperatures. Overall it is not a large effect but is one that needs to be addressed during licensing; it is considered of medium regulatory concern. The maturity level in knowledge of this phenomenon mirrors the distribution in the data: high below 10 at% burnup, medium above 10at % burnup, and low above 20 at%. Early work on this area was performed by Meyer (1974).

Fuel Cladding Chemical Interaction (FCCI): FCCI is an in-reactor phenomenon once feared as Life-limiting in SFR stainless-clad MOX fuel elements. It is caused by the radial migration to the fuel-cladding interface of fission products cesium, iodine and tellurium, and oxygen freed by fission. Oxidative corrosion of the steel inner surface in the presence of fission products can then occur, either as a uniform reaction, or—under the right conditions—as attack along grain boundaries of the steel which have become denuded of $Cr_{23}C_6$ precipitates in-reactor, i.e., thermally sensitized.

Uniform FCCI can be considered as simple wastage or thinning of the cladding; grain-boundary FCCI is less predictable and potentially could lead to cladding failure, although none has ever been observed. It was discovered, however, that FCCI can be largely suppressed by lowering the initial stoichiometry of the fuel; oxygen-to-metal ratios of 1.94-1.95 will inhibit occurrence of FCCI to well beyond 10at % burnup. The phenomenon was much studied in the 1970s and 1980s, and is considered to be well understood; it has also been extensively reviewed(Lawrence et al., 1990). For these reasons FCCI is designated of medium regulatory concern. The maturity level in knowledge is high up to 10at % burnup, medium to 20 at% burnup, and low above 20 at% (post-irradiation examination of the CDE MOX elements from FFTF should soon alter the latter designation).

Fuel-Coolant Compatibility: Fresh UO_2 and $(U,Pu)O_2$ react directly with sodium to form sodium uranate (Na_3UO_4) or sodium urano-plutonate (Na_3MO_4, where M = U,Pu). The reaction takes place in the fuel-cladding gap and in open porosity and cracks in the fuel up to about 1000^oC (the dissociation temperature of the reaction products). The reaction products have about half the density of the fuel they replace, so that local swelling occurs and a cladding breach may be extended. For medium burnup fuel, in which cesium uranate or cesium urano-plutonate has already formed at the fuel surface, the reaction is different: the chemically more active sodium replaces the cesium in the reaction phase and cesium is lost to the primary coolant; for these conditions local swelling is much less, although with brittle cladding a breach may still propagate.

Fuel-coolant incompatibility is thus considered a Life-limiting phenomenon of high regulatory concern; there is no analogous phenomenon for metallic fuels, which are entirely compatible with sodium. However, a 15-year program of run-beyond-cladding-breach (RBCB) testing of MOX fuel elements in EBR-II (Lambert et al., 1990) cast an interesting light on the phenomenon. Provided the swelling from fuel-sodium reaction does not cause unstable splitting of the cladding, the fuel-sodium reaction product, once formed, becomes a barrier (a “scab”) to further reaction so that continued operation with the failed fuel is possible for several months, including shutdowns and startups. Because of experience from this RBCB program, knowledge of the phenomenon is considered high to 10 at% burnup, decreasing at burnups above 10 at%.

IV.C.2 Thermophysical Properties

Table 10 lists the thermo-physical properties of importance in analyzing oxide fuels for licensing purposes. Of highest regulatory importance are those properties supporting thermal analysis of the fuels under irradiation—thermal conductivity, heat capacity, thermal expansion, and melting temperature. Other thermo-physical properties are of lesser (medium or low) importance.

Being a universal fuel for LWRs, UO_2 has been thoroughly studied, particularly for its thermal conductivity. Recent compilations of thermo-physical properties of UO_2 were published by the IAEA (1997), and stored in the THERSYST system at Stuttgart University; and by the NRC (2011), and stored in the MATPRO database maintained for the U.S. industry. The IAEA report also included SFR MOX fuel. Among many other reviews, the most succinct comparison of the thermophysical properties of UO_2 and MOX was given by (Carbajo et al., 2001).

It was concluded by the Gap Analysis experts that the thermo-physical properties of both UO_2 and MOX fuels have already been sufficiently well determined to be used immediately in support of the licensing of an oxide fueled SFR.

IV.C.3 Minor Actinide Additions

Table 3 indicates the minimal U.S. experience with SFR oxide fuel containing minor additions of actinides. Some irradiation tests have been performed in the ATR reactor in Idaho and in the Phenix reactor in France. Although PIE of these tests is still in progress, initial results are encouraging (Hayes, 2011). Work has also been performed in the JOYO reactor by JAEA to study the effect of minor actinide additions on the thermal conductivity of MOX fuel. Again, results were encouraging—Am additions of up to 3 wt% only slightly reduce the thermal conductivity of MOX fuel (Morimoto et al., 2008). Nevertheless, it is clear that significantly more work is required before any licensing case could be made for MOX fuel containing actinides.

IV.C.4 Fission Product Carry-over

Fission product carry-over is really only a concern for metal fuel that has been recycled by an electrochemical process, wherein fission-product lanthanides can be carried over into the re-fabricated fuel. If the Purex process is employed for oxide recycle this is not a real concern for any of the fission products.

It should be noted that both the PFR and Phenix SFRs have been operated with reprocessed MOX driver fuel obtained via the Purex route. No deleterious effects attributable to fission-product carry-over have been reported.

IV.D Cladding and Duct Materials

At the current time only two alloy classes have enough radiation response data to seriously consider them as structural materials for a licensable reactor. The first alloy class is austenitic stainless steel (Type 316 or D9). This steel is limited to doses of ~100 dpa (and therefore the fuel burn-up to 10-11at%) due primarily to embrittlement concerns arising from void swelling and secondarily from both linear and volumetric distortion induced by swelling and irradiation creep. For such exposure limits there are essentially no significant gaps in required data and knowledge.

The second alloy class contains ferritic and ferritic-martensitic steels, especially those in the 12Cr category. These steels are much more resistant to the onset of void swelling but suffer from loss of creep strength at the higher temperatures required for efficient power generation. HT9 is the U.S. candidate and has been used to successfully build and operate without failure a set of fuel assemblies in FFTF. If this alloy is used within the boundaries of the FFTF experience there are no significant gaps in required knowledge. Such boundaries are probably insufficient for efficient power generation, however.

There is currently a lot of attention being paid to oxide dispersion strengthened ferritic-martensitic alloy in order to retain both swelling resistance and high temperature creep strength at the same time. Given the very limited data on radiation response of these alloys it is premature to consider these steel seriously for a licensable fast reactor and therefore a gap analysis is also premature.

IV.D.1 Introduction

The structural alloys used in any fast reactor can be grouped into three major categories describing their function, each with a different set of limitations and each with a different set of potential gaps in knowledge of the needed properties. In order of increasing severity of nuclear, thermal and chemical environments that the alloy will experience, these categories are the out-of-core structural components, the ducts that contain the fuel assemblies, and the fuel pin cladding, with the latter including the wire wrap used to separate the pins.

Any steel used as structural components in liquid metal cooled fast reactors must withstand an exceptionally strenuous and challenging environment, even in the absence of neutron irradiation. Depending on the particular fast reactor concept, the inlet temperature during reactor operation can range from ~250°C to ~400°C, although the start-up and shut-down sequences of these reactors sometimes utilize lower standby temperatures. The maximum temperature can range as high as 650–700°C for some components, although most non-fueled components reach maximum temperatures in the range of 400–560°C. During operation the steel must also withstand the corrosive action of fission products on some surfaces and flowing liquid metal coolant on other surfaces. Dependent on the nature of the component and the length of its exposure, there may also be significant and time-dependent levels of stress acting on the component.

Most importantly, however, the steel must survive the macroscopic consequences of a continuous microstructural alteration that arises from atomic displacements arising from collisions of energetic neutrons with atomic nuclei. The driving forces for this alteration are primarily displacements of atoms from their lattice sites, measured in units of displacements per atom (dpa), and secondarily from transmutation to both gaseous (He, H) and solid transmutants. Reviews of these processes and their consequences in austenitic stainless steels are presented in refs. (Garner, 1994) (Garner, 2010) (Garner, in press).

Neutrons in fast reactors have a spectrum of energies that is dependent on the reactor type, core loading, coolant and fuel type. The most energetic neutron spectrum will be found in fast reactors with heavy metal coolant such as Pb-Bi. The mean neutron energies in the center of a sodium-cooled core are on the order of 0.8 MeV in metal-fueled reactors and 0.45–0.55 MeV in mixed oxide-fueled reactors. The lower mean energy of the latter fuel type reflects the better neutron moderating ability of oxygen in the oxide fuel. These spectra produced approximately 5 dpa per 10^{22} n cm^{-2} (E > 0.1 MeV) in the center of EBR-II and 4.2–4.6 dpa per 10^{22} n cm^{-2} in various core loadings of FFTF. This in turn corresponds to peak atomic displacement rates on the order of 10^{-6} dpa/s. When using dpa as an exposure parameter for the structural alloy we need not be concerned by the type of coolant or the type of fuel (metal or ceramic) inside of the cladding.

The radiation-induced microstructural evolution of the steel leads to changes in physical properties such as elastic modulii and thermal conductivity, but also causes very pronounced changes in mechanical properties. Even more importantly, however, new forms of dimensional instability arise. In increasing order of significance are strains arising from radiation-induced segregation and precipitation or dissolution of precipitates, larger strains arising from radiation-enhanced creep (by orders of magnitude larger than thermal creep at lower temperatures), and finally, void swelling in which large increases in volume can occur.

The latter arises from the formation of vacuum-filled, crystallographically-faceted cavities that eventually come to dominate the microstructure. Examples of both microscopic and macroscopic consequences are shown in Figures 2 and 3. The onset of void swelling is very sensitive to metallurgical starting state, composition, irradiation temperature, dpa rate and helium generated by transmutation. The onset of swelling is somewhat less sensitive to applied stress.

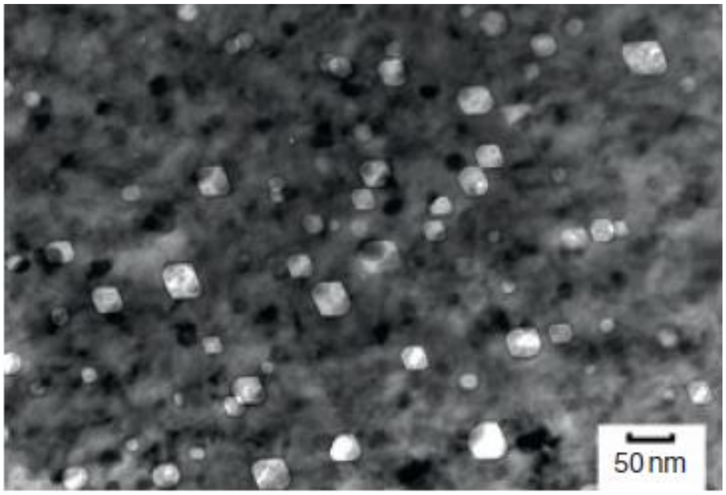

Figure 2. Void swelling and M23C6 carbide precipitation produced in annealed 304 stainless steel after irradiation in the EBR-II fast reactor at 380°C to ~22 dpa (Garner, 2002).

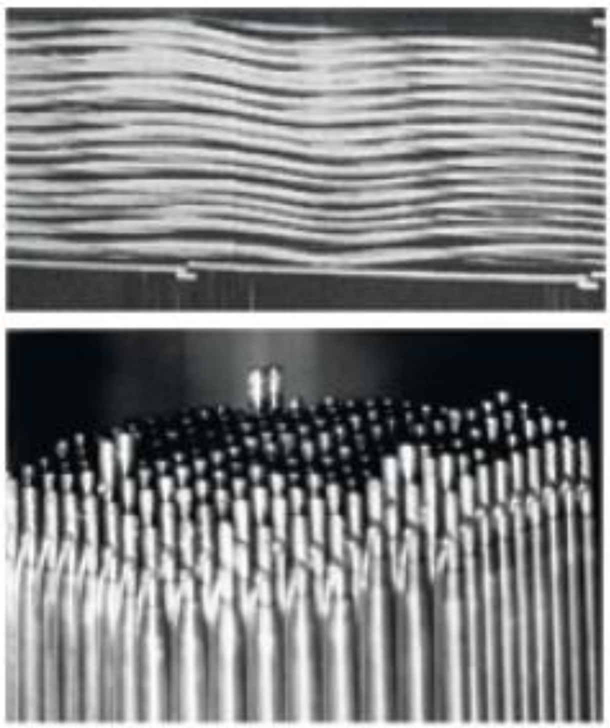

Figure 3. (top) Spiral distortion of AISI 316-clad fuel pins induced by swelling and irradiation creep in an FFTF fuel assembly; (bottom) Swelling-induced changes in length of fuel pins in this assembly in response to gradients in temperature, dpa rate and production lot variations (Makenas, Chastain and Gneiting, 1990).

IV.D.2 Problems associated with irradiation creep and void swelling

Problems arising primarily from irradiation creep can be mitigated somewhat by design considerations such as gas plenums, thicker cladding, larger flow spaces, intermediate supports restraining long components, etc., but swelling and its consequences are more difficult to mitigate.

Void swelling was found to be the Life-limiting phenomenon in austenitic stainless steels in fast reactor application. Working in conjunction with irradiation creep, tremendous distortions and volume changes can be produced by swelling, although to some extent these distortions can be accommodated in the design process if they can be accurately predicted in advance. For every design, however, there is some Life-limiting swelling limit that can be tolerated, sometimes based on closure of flow channels but often on other factors such as interference with movement of safety rods or development of unacceptable withdrawal forces.

The latter is particularly important in that void swelling when passing beyond ~10 at% increase in volume leads to development of a severe form of embrittlement in austenitic steels whereby there is total loss of elasticity and the tearing modulus of austenitic steel goes to zero. Voids at >10% swelling so modify the microstructure and compositional distribution that austenitic stainless are driven toward a martensite instability from which there is no return. This new form of embrittlement in effect becomes the Life-limiting criterion and poses a safety issue of relevance to responsibilities of USNRC. An extreme example of such embrittlement is shown in Figure 4 where ducts of three assemblies in BOR-60 failed due to high withdrawal loads arising from swelling-induced bowing and fattening of the ducts. It is of particular significance that the duct failed since ducts operate at lower temperature than fuel pins and as a consequence generally swell less.

It is difficult to preclude large levels of void swelling at higher dpa levels unless the reactor is operated at very low temperatures (<300°C) so the emphasis has been on developing an understanding of swelling and then optimizing the compositional, fabricational and environmental conditions of the steel. In general, the path chosen in most national programs was to develop a "D9-type" steel, with increases in Ni, Si. P, Ti and other elements (relative to AISI 316) to delay the onset of swelling, especially when combined with cold-working in the range of 20-25%. Eventually, however, all steels will start to swell at ~1%/dpa at all operating conditions of relevance to fast reactor operation.

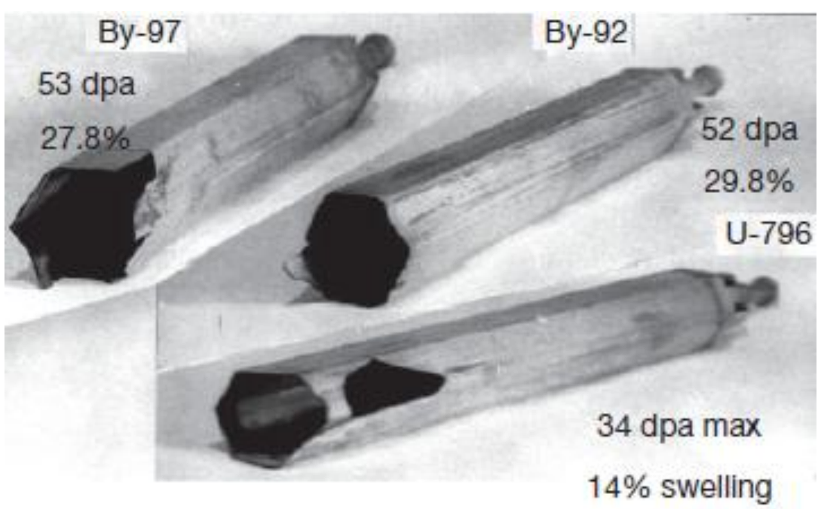

Figure 4. Severe embrittlement and failure of three BOR-60 reflector assembly ducts made of annealed X18H10T, the Russian equivalent of AISI 321 stainless steel (Neustroev et al., 2000).

Failure arose from high withdrawal loads arising from swelling and bending, the latter in response to flux gradients across the assemblies.

Another very significant problem with void swelling is that limits imposed on maximum exposure limit the burn-up of the fuel, strongly impacting the economics of the reactor. With 316 Stainless Steel and mixed oxide fuel having typical enrichments, maximum burn-ups of 10-11 at% were reached in the U.S. LMR program. Similar upper limit values were reached in other national programs. Although under some conditions steady-state swelling can be postponed until ~150 dpa, a more practical and dependable limit is ~100 dpa, especially since various of-normal histories can abruptly end the incubation regime of swelling. Such a burn-up limit is not a show-stopper for fast reactors if used for power generation. BN-600 for instance has successfully and economically generated power for decades using austenitic steels as cladding. However, if the fast reactor is envisioned to serve another purpose such as transmutation of actinides, then austenitic steels are not adequate.

If a burn-up limit of 10-11at% is accepted for power generation in the U.S. then the 316 stainless steel-clad, mixed oxide or metal fuel technology is a mature technology within the U.S. and there are very few gaps in required knowledge to be filled. Other non-swelling-related issues such as sodium compatibility and fuel-clad mechanical interaction are suitably understood and under control. However, if higher burn-up is required, then lower-swelling alloys compared to stainless steel are required, and therefore exposures greater than ~100 dpa must be sought before steady-state swelling begins.

IV.D.3 Other possible alloy classes to reduce swelling

Extensive national programs in U.S., U.K., Japan, France, Germany and the Soviet Union have all explored various categories of other alloys. With only relatively minor differences of composition chosen, all programs have reached roughly the same general conclusions.

High nickel alloys, both solute-strengthened and precipitate-strengthened offer longer delays before swelling goes into steady-state but are accompanied by new forms of embrittlement, especially involving the formation of brittle phases that coat the grain boundaries. Additionally higher levels of helium and hydrogen form as a result of the higher nickel content and appear to contribute additionally to grain boundary embrittlement. Several national programs, including in the U.S. but especially in the U. K., invested considerable research and development on this class of alloy but eventually abandoned this approach.

The use of more exotic materials, especially refractory alloys, were found to be inherently brittle after irradiation and were therefore found to be completely unsuitable. Only ferritic and ferritic-martensitic alloys offer the required superior swelling resistance, primarily because the bcc crystal structure is more resistant to void swelling, with an apparent steady-state swelling rate of ~0.2%/dpa, approximately one-fifth that of the austenite fcc structure (Garner, Toloczko and Sencer, 2000). Additionally, the incubation dose for swelling in bcc steels is much larger, partially due to the crystal structure, but also because the absence of nickel in these alloys significantly reduces the transmutation-production of helium to enhance stability of void nuclei. The most comprehensive reference on these alloys and their radiation response is contained in (Klueh and Harries, 2001).

Most national programs have focused on ferritic and ferritic-martensitic alloys, and national favorites have emerged in all countries. For the purpose of this analysis we will focus on only two alloys, both produced in the fully-tempered condition.

The first alloy is EP-450, a ~12Cr duplex alloy (~50% ferrite and 50% tempered martensite) extensively used in countries of the former Soviet Union (Bibilashvili et al., 2005). Ducts surrounding austenitic-clad fuel pins in BN-350 and BOR-60 are routinely made from this alloy.

EP-450 has the largest, best-documented and most varied data base of any alloy. It is cited here as an example of the promise of ferritic-martensitic alloys in general, best demonstrated by reaching ~160 dpa in fuel assemblies in BOR-60 (Povstyanko et al., 2010). Void swelling was still in the incubation regime without hint of steady-state swelling when this experiment was terminated. Recent ion irradiation studies confirm that the eventual steady-state swelling rate in EP-450 is the previously predicted ~0.2%/dpa, however, but will not be obtained before ~200 dpa has been reached (Voyevodin et al, 2011).

The second alloy is HT9, the U.S. candidate alloy for fast reactor application, produced in the fully tempered condition. Its primary attractiveness is that it was used to construct the Core Demonstration Experiment (CDE), a mixed-oxide sub-core of fuel assemblies irradiated in the center of FFTF, which yielded outstanding results for clad, wire and duct (Laidler and Jackson, 1990) (Leggett and Walters, 1993). Maximum swelling of only ~0.3% was reached in one examined duct at ~155 dpa at ~440°C (Sencer et al., 2000). The cladding appears to have swelled less by virtue of its higher temperature but examination is still in process.

Pressurized tubes of HT9 irradiated in FFTF at ~208 dpa and ~400°C swelling at 0.9-2.1%, however, increasing with stress level (Garner, Toloczko and Sencer, 2000). Usually, stress-enhancement of swelling signals the end of the transient regime of swelling so the transient regime at 400°C should be assumed to be on the order of approximately 200 dpa.

However, it should be noted that it was necessary to drop the power level of FFTF from 400 MW to 280 MW to accommodate the CDE activity. Ferritic and ferritic-martensitic steels do not maintain their strength at increasing temperatures in a manner comparable to those of austenitic steels. Therefore it was necessary to reduce the outlet temperature of the fuel assembly. Otherwise the fuel pins will fail as fission gas pressure builds up inside the pins.

All other relevant issues such as corrosion, fuel-clad interaction, etc. appeared to be well in control as evidenced by the successful operation without any failure. However, the CDE sub-core, though probably capable of reaching higher exposure, was terminated largely due to programmatic changes. Therefore its upper limit of HT9 subassemblies was not reached.

If future reactor designers are content to stay within the dose-temperature parameter space explored by the CDE sub-core then there are no significant knowledge gaps. Such a limitation on assembly outlet temperature would severely limit the economic viability of a power-generating plant, however, making it unlikely that licensing would be sought for a plant of that design.

IV.D.4 Oxide dispersion strengthened alloys

There is currently a significant effort in various national programs in Europe, Asia and the U.S. to extend the operating temperature range of ferritic-martensitic steels to higher temperatures while still retaining the lower swelling characteristics of ferritic-martensitic steels, not only for light water cooled reactors but also for fusion and spallation-driven devices. A number of studies have shown that in the absence of neutron exposure, finely dispersed particles of various metal oxides delay the onset of accelerated thermal creep to temperatures relevant to efficient power generation (Odette et al., 2008) (Odette and Hoelzer, in press) (Alinger, Odette and Hoelzer, 2009) (Ohnuma, Suzuki, and Ohtsuki, 2009) (Miller, Russell, and Hoelzer, 2006) (Sasasegawa, et al., 2009). One ODS alloy, MA957, was shown to retain its creep resistance in the presence of neutron exposure in FFTF (Toloczko et al., 2004).

The oxide phases can be introduced by mechanical alloying but usually result in highly textured microstructures with anisotropic properties. As a consequence it has been found that it is very difficult to manufacture tubing and to weld it into fuel pins. Newer approaches focus on growing nano-oxides in place as one way to reduce anisotropy and its consequences.

For the purposes of this gap analysis, however, it is considered to be premature to consider this class of alloys for serious near-term application. Most promising alloys have little or no irradiation data. Therefore a gap analysis is not relevant to this alloy class.

On Table 5 for cladding and Table 8 for ducts it is seen that for HT-9 the technological base is mature up to about 100DPA but additional data is required beyond 100DPA.

On Table 7 and Table 8 for 316 and HT9 for duct phenomena, respectively, data may exist for bundle, bundle-duct, and duct-duct interactions but it was uncertain if the data has been preserved in a useable form.

Table 6 shows that the technological base for 9Cr1Mo (T91) is not mature even though it remains a good alloy possibility. Additional data likely exists in France and Japan but its availability is uncertain.

IV.E Fuel Performance Codes

The following sections provide a summary of steady state and transient fuel performance codes.

IV.E.1 LIFE-METAL

The LIFE-METAL fuel performance code (Billone et al., 1986) (ANL-IFR-169, 1992) has been developed to predict the behavior of metallic fuel pins in fast reactors environment as a function of reactor operating history. The code has evolved from the LIFE series of codes (Jankus and Weeks, 1972) which perform steady-state and design-basis-transient analyses for the thermal, mechanical, and irradiation behavior of nuclear fuel pins. The original code was developed for UO2 and mixed oxide fuels for use in fast reactor systems where LIFE-4 Rev .1 is the latest oxide fuel version of the code (Boltax et al., 1990). Another version of the code is LIFE-4CN (Liu, Zawadzki and Billone, 1979), which was the basis for LIFE-METAL, and included two fuel options ((U, Pu)C and (U, Pu)N). All code versions include detailed thermo-mechanical analysis that is performed in the radial direction with provisions to specify up to 20 radial rings for the fuel/cladding system, where different rings are used for thermal and mechanical analysis.

Axial variations in operating conditions are accounted for by using powers and fast fluxes for up to nine fuel axial nodes and one plenum node. Thermally, the axial nodes are coupled through the calculated coolant temperatures. Axial heat conduction is ignored and there are no provisions for mechanical coupling between axial nodes. A detailed mechanical analysis is performed for both fuel and cladding utilizing the generalized-plane-strain assumption for each axial segment and incorporating a large strain capability. The solution procedure involves iteration on local total strain within each time step, and the solution procedure is explicit in time.

LIFE-METAL code development has been associated with the Integral Fast Reactor (IFR) program (Chang, 1989) where the code was the focus of the program activities related to prediction of fuel-pin behavior under normal operating conditions. Predictions of interest to the nuclear design are changes in fuel length and fissile content due to burnup and breeding. Thermal predictions of fuel temperature, design margins to fuel melting, and design margins to low-melting-temperature alloy (e.g., U-Fe) formation are also of interest. Mechanical predictions useful to designers are fuel-cladding mechanical interaction (FCMI) and fuel-cladding chemical interaction (FCCI), cladding deformation and design margin to significant coolant flow area reduction, and cladding damage and design margin to cladding failure due to fission gas pressure loading.

LIFE-METAL Validation

The following validation discussion is based on the last code validation activities performed by Billone (Billone, 1994). This validation effort has been extensive as it used post irradiation examination (PIE) data that are available from a large number of metallic fuel-pin irradiations at EBR-II and FFTF (Crawford, Porter and Hayes, 2007). Post irradiation examination (PIE) data include fission gas release, fuel volumetric and fuel length change, cladding diametral change, and cladding wastage. Axial profiles are available for fuel radial growth at low burnup (prior to and including initial fuel-cladding contact) and for cladding radial growth for a wide range of burnups and fast fluences. Some data that are available on a more limited basis are radial and axial variations in U, Pu and Zr content, fission gas porosity, axial variations in fraction of porosity filled (logged) with Na; and depth of C-depleted and Ni-depleted zones in HT9 and D9, respectively. Fairly complete sets of data are available for 80 fuel-pin irradiations (111 pins in total were used in the validation). Limited data (e.g., fuel length change, cladding diameter change) are available for hundreds of irradiated fuel pins.

The validation database includes three cladding types (cold-worked, austenitic D9 and 316 stainless steels and HT9 ferritic/martensitic steel) and eight fuel compositions (U-10Zr, U-3Pu-10Zr, U-8Pu-10Zr, U-19Pu-6Zr, U-19Pu-10Zr, U-19Pu-14Zr, U-22Pu-10Zr and U-26Pu-10Zr, where the numbers represent weight percents). The data from the 111 pin irradiations fall into one or more of the following categories: fission gas release, fuel axial strain, fuel diametral strain, cladding diametral strain and penetration depth (wastage) at the cladding inner diameter due to ingress of fission products and egress of cladding constituents. For the last three categories, axial profiles are often available. This implies a large number of data points per fuel pin irradiation. Also, in the case of fuel axial expansion and peak cladding strains, which are routinely measured for all elements within a subassembly, the number of data points is much larger than the number of validation cases.

LIFE-METAL Status

The latest calibration of the LIFE-METAL code was performed just before the termination of the IFR project in 1994 (Billone, 1994). Sets of verification test problems that correspond to data from different EBR-II experiments are available and have been used systematically to verify the code calculations.

Minor changes have been done to the code since its calibration. Those changes did not affect the code's calibration and were mainly aimed at correcting a code error associated with FCCI for fuels with long irradiation periods. Since its last validation activity, the code has been used in a few occasions to support the evaluation of metallic fuel designs associated with advanced fast reactors designs such as the 4S and ARC reactors (Yacout, Tsuboi and Ueda, 2009). Currently, the code has limited number of users and is not released to the national code center as it needs detailed documentations and re-validation effort to release it. Further, calibration and validation effort of the code can be done once further data from other EBR-II experiments are generated as part of efforts to create a database for metallic fuel irradiated at EBR-II.

LIFE-METAL Limitations

The code lacks the implementation of mechanistic models in a good part of its development. Thus, the code is limited in extrapolating fuel performance outside of the validation range of parameters, since a lot of the code models are based on correlations rather than mechanistic models. However, the code still can be useful for scoping calculations outside of its range of validation. As stated previously, the thermo-mechanical modeling part of the code has limitations due to the axial nodes being thermally coupled only through the calculated coolant temperatures, axial heat conduction being ignored, and there being no provisions for mechanical coupling between axial nodes. Finally, there are no models in the code that are relevant to evaluations of transuranics bearing fuel.

IV.E.2 LIFE-4 (oxide)

The LIFE-4 (Rev. 1) code (Boltax et al., 1990) was developed to calculate the thermal and mechanical behavior of mixed oxide fuel elements in a fast-reactor environment. The code is the reference national code for modeling the thermal, mechanical and materials performance of fast reactor oxide fuel and blanket pins during normal operation and during transients up to cladding breach. It integrates a broad material and fuel-pin irradiation database into a consistent framework for use and extrapolation of the database to reactor design applications. It can also be used in the design and analysis of transient fuel-pin experiments, and in the identification of critical experimental areas. The code has a one-dimensional generalized-plane-strain mechanical analysis procedure for fuel ([U,Pu] O_2 and UO_2) and cladding (several types including different types of 316 stainless steel, HT9, and D9). It has a steady state and transient thermal analysis system for fuel, cladding and various flowing or static coolants. Thermal and mechanical behavior including thermal expansion, fuel restructuring, cladding deformation, and cladding breach is calculated from a number of phenomenological models. The thermal analysis includes the following:

- Establishing fuel and cladding temperatures

- Calculating stress-strain independent, but temperature dependent, diffusion phenomena such as:
    - Pore migration
    - Grain growth
    - Oxygen migration
    - Pu migration
    - Fission gas release.

The mechanical analysis is a stress-strain calculation in the fuel and cladding and includes the following:

- Elastic creep (irradiation and thermal creep)
- Plastic flow
- Swelling
- Thermal-expansion strains.

The code also includes wastage model, and it takes into account the fuel cladding mechanical interaction in calculating the cladding diametral strains.

Similar to the LIFE-METAL code, the fuel pin is divided axially into a maximum of nine fueled sections and one plenum section. Each axial section is divided radially into a maximum of 20 cylindrical rings for mechanical analysis. Each axial node is mechanically independent of all other axial nodes. Thermally, the axial nodes are coupled through the calculated coolant temperature. Axial "lock-up" effects or similar axial coupling effects are not modeled. Thus, if a given pin section was run as part of a one-node problem or as part of a five-node problem, the computed temperatures, stresses, etc. would be the same except for two calculated parameters which are:

- The plenum pressure, which will be different because of differences in the total amounts of fission gas released.
- The axial vapor-phase transport, where the differences are very small for long (~ 1 meter) pins.

Because of this axial independence, each axial node can be solved separately. Finally, the code has a special option for blanket fuel pins that are characterized by larger diameters.

LIFE-4 Validation

LIFE-4 (Rev. 1) has been validated against pin data from 64 pins that were irradiated under steady-state conditions in EBR-II, 12 pins that were transient-tested in the TREAT reactor, and 13 pins that were irradiated in FFTF. Data include both (U,Pu)O_2 mixed-oxide fuel pins and UO_2 blanket pins which were irradiation tested under steady-state and transient conditions. For fuel pins under steady-state conditions, this calibration/validation covered the following ranges of operating conditions: 16-50 kW/m (peak power), 1.5-5x10^{15} n cm^{-2} s^{-1} (peak flux), 360-700 ^{o}C (peak cladding temperature), 0-13% burnup and 0-1.5x10^{23} n.cm^2 (peak fluence). The code calibration was done by adjusting the less well-defined parameters in the fuel models and properties within physically realistic limits or measurement uncertainties to minimize the deviations between code results and fuel pin post-irradiation examination (PIE) data. The beginning-of-life thermal calibration was performed by adjusting the high temperature end of the fuel thermal conductivity equation, the accommodation coefficient in the gap conductance model and the constants in the pore velocity expression. For the later-in-life thermal calibration, changes were made to the dependence of fuel thermal conductivity on burnup (due to solution of fission products). The primary parameters employed for the mechanical calibration were constants in the fission gas release expressions and the fuel swelling model. The key fuel pin data used were fission gas release and cladding mechanical strains. Also, several other fuel constants were changed during the course of calibration to achieve improvements in code predictions not obtainable by varying the parameters described above.

The code validation was performed by investigating code predictions for fuel pins, which were not used in the calibration of the code. No adjustments were made to code calibration constants for the analysis of the validation pins. This is a summary of the validation data:

- The cladding for all pins except the high burnup F20 pins was 20%cold-worked AISI Type 316 stainless steel of the N-lot type.
- The high burnup F20 pins used solution-annealed AISI Type 316 stainless steel cladding.
- All pins are from mixed fuel pin tests, except the W20 pins, which are from blanket pin tests.
- All pins except DEA-2 pins were irradiated in EBR-II. DEA-2 pins were irradiated in FFTF.

LIFE-4 Status

The LIFE-4 (Rev.1) version of the code is well documented and maintained through the national code center. This version dates to the early 1990's and is the latest code version. There have been further code validation efforts and modifications using FFTF experimental data that were not available during Rev.1 version. We do not have access to documentations of this but it might become available through the recent effort to preserve FFTF documents and codes. Again, the code does not have any models specific to transuranics fuels.

LIFE-4 Limitations

There are a number of limitations on the code ability to predict various features of the actual behavior of fuel pins, which are caused by errors in, or lack of, models of fuel and cladding behavior. The following are examples of important code short-falls, which are mainly related to the limitation, that were imposed on the code developers in the past, mainly in order to reduce the computation time, in addition to inherent use of correlations in some parts of the code compared to mechanistic models. For example, the axial mechanical analysis in the code is a simple one, which assumes that no axial section of fuel is affected by the condition or behavior of any other axial section. That is, axial lock-up effects are not taken into account. This leads to mechanical diametral strains near the bottom of the fuel column that tend to be under-predicted. The same strains near the top of the fuel column tend to be over-predicted.

Another consequence of the simplicity of the code's mechanical analysis is that the thermal and mechanical analyses in the code are primarily coupled through the gap-conductance model. Both analyses are not solved simultaneously; instead, the thermal analysis uses the fuel-cladding gap or contact pressure from the mechanical analysis of the previous time step. Consequently, the code has a tendency to become unstable and produce oscillating solutions under certain conditions. The code is limited in use to oxide or mixed oxide fuels without transuranics.

IV.E.3 FPIN2

FPIN2 (Hughes and Kramer, 1986) (Kramer et al, 1992) code has been developed to model the thermal and mechanical behavior of IFR metallic fuel pins. It was developed primarily to analyze the behavior under accident transient conditions. The emphasis in the development and validation of the code has been the incorporation of models relevant to the time scale and temperature range of accident transients. It requires the user to provide data describing the pre-transient condition of the fuel pin. FPIN2 has relied on TREAT tests as the primary source of data for overall code validation. The regime of extended transients requires an extrapolation of FPIN2 to longer times. It has been also applied to the Whole Pin Furnace tests conducted at Argonne as part of the IFR program. There are many similarities between LIFE-METAL and FPIN2 in the fundamental assumptions that determine the governing equations that are solved. For instance, both codes use a finite difference formulation of the heat transfer which assumes that heat is only conducted radially in the fuel and the cladding and convected axially by the coolant. For the purpose of analyses of experiments such as the Whole Pin Furnace tests, options are available to bypass the coolant calculation and input directly the known cladding surface temperatures. The mechanical analyses in both codes assume axial symmetry and generalized plane strain which results in an essentially one-dimensional (radial) calculation for each axial node of the fuel pin. LIFE-METAL uses a finite element formulation of the governing equations whereas FPIN2 uses a finite pin formulation. In addition, both codes use similar fundamental properties of metallic fuel pins. As mentioned above, differences in the regimes that the LIFE-METAL and FPIN2 address have resulted in major differences in the models contained within the codes. To some extent these differences have also influenced the formulation of the equations and solution procedures (e.g., finite difference vs. finite element). One of the primary differences in methodology that affects the calculations is how the code determines the deformation and failure of HT9 cladding. Here, LIFE-METAL has based the models on long-term creep tests, while FPIN2 has based the models on tensile test data and FCTT (Fuel Cladding Transient Tester experiments (Cannon, Huang and Hamilton, 1991) data.

IV.E.4 SEIX3

The SIEX3 code (Baker and Wilson, 1986) was developed at Hanford Development Engineering Laboratory (HEDL) during the project to design, construct, and operate the Fast Flux Test Facility (FFTF). The code is strongly based on experimental data obtained from irradiations performed in EBR-II, and in FFTF. The code contains correlated models based on experimental data that describe thermal performance and phenomena causing dimensional changes. The code predicts cladding damage and failure due to stress rupture in MOX fuel pins irradiated at steady-state conditions in an LMR.

In SIEX3 modeling, the pin of interest is analyzed using the driver fuel pin correlations if the fuel Pu/(Pu+U) ratio is greater than 0.12 and fuel pellet diameter is less than 0.32 inch; otherwise it is assumed to be a radial blanket fuel pin and is analyzed using the blanket fuel pin correlations. In either case, the fuel column is divided into a user-specified number of axial segments (up to 21) of equal length. One-dimensional radial heat transfer is calculated at the axial center of each segment. The heat generated is integrated over the axial segments to calculate the coolant temperatures along the pin, from input values of axially varying linear power, coolant mass flow rate and inlet temperature. The temperature-drop across the cladding wall includes the power in the cladding due to gamma heating (0.7% of total power). The following are calculated for each segment:

1. Coolant temperature
2. Cladding outer and inner surface temperatures
3. Fuel-cladding gap size
4. Movement of void volume from the fuel-cladding gap to the central void in the fuel
5. Solid fission product (gray phase) buildup in the fuel-cladding gap
6. Fuel-cladding heat transfer coefficient
7. Fuel radial temperature distribution
8. Fuel restructuring radii for equiaxed and columnar zones
9. Displacement of the fuel and cladding caused by swelling and thermal expansion
10. Fission gas generation and release
11. Cladding wastage, stress, creep and damage.

The cladding performance calculations in SIEX3 are based on the observed performance of reference design liquid metal reactor fuel pins with a fuel smear density of 85% TD. The calculations assume that pin diameter changes depend primarily on neutron-induced swelling and cladding creep strain caused by fission gas pressure loading only. SIEX3 calculates the cladding thermal expansion, cladding swelling, wastage, thermal creep, irradiation creep, and stress rupture damage [cumulative damage fraction (CDF)].

The code uses two time steps, one defined by the code user for writing output results, and the other of two effective full power days (EFPD) for carrying out mechanical calculations. This short time step is used to ensure accuracy of calculations that are sensitive to a small change in fuel pin fluence, stress, or burnup.

SIEX3 has the following important capabilities:

1. The code allows annular fuel. The fuel pellet central void model in the code accounts for two processes:
    a. Densification of the columnar fuel grain region, and
    b. Porosity movement from the fuel-cladding gap to the central void, due to fuel cracking and subsequent healing.
2. The fabricated fuel oxygen-to-metal ratio is an input to the code. This ratio is used in correlations for columnar grain growth temperature, and equiaxed grain growth temperature.
3. For rapid startup with fresh fuel, the code uses a special columnar grain growth temperature, in addition to that for normal irradiations.
4. The code accounts for void volume caused by fabricated end dishes in the fuel pellets.

5. The code provides a single convenient input parameter to specify the recommended cladding performance models for twelve cladding materials.
6. For modeling radial blanket fuel pins, the code uses correlations specifically developed from the irradiation test database for blanket pins.

SIEX3 Status and Limitations

The version 3 of the code, SIEX3, is the latest version of the code available through the national code center. No further developments are done with the code.

SIEX3 has the following important limitations:

1. The code uses a single input value of the external pressure acting on the cladding outer surface. It does not account for the axial variation of the coolant pressure acting on the cladding outer surface.
2. There is no provision to model a fuel pin fission gas plenum located below the fueled section of the pin.
3. The code does not compute and print the change in fuel column length.

IV.E.5 FEAST

FEAST-METAL and FEAST-OXIDE are fuel performance codes developed for predicting steady state and transient behavior of U-Pu-Zr metallic fuel alloys and mixed oxide fuels with stainless steel clad in sodium fast reactor environments. The codes are developed at MIT with support from NRC. The code properties and validation databases are derived from information available in the open literature for both types of fuel. Other validation and calibration data used with the LIFE series of codes were not used in the validation of FEAST, so it has a limited validation database. The thermo-mechanical models in the code are similar to those in LIFE codes models, while the fission gas release models used in FEAST are mechanistic rate equations based models compared to the correlations used in the LIFE codes.

IV.E.6 DEFORM-4

The DEFORM-4 model was developed at ANL as part of the SAS4A safety analysis computer code system. DEFORM-4 contains detailed phenomenological models of MOX fuel behavior, including fission gas generation and release, porosity migration, fuel and cladding swelling, and fuel-cladding mechanical interactions. The models in DEFORM-4 are coupled and integrated with the fuel pin heat transfer, coolant dynamics, and material melting and relocation models in SAS4A. The modeling in DEFORM-4 has been upgraded by a German-Japanese-French consortium (Imke, Struwe and Pfrang, 1995) to reflect fuel materials and experimental data generated in their national LMR development programs. The upgraded model is designated DEFORM-4.

IV.E.7 Knowledge preservation and database development (Metal fuel performance)

Most of the existing knowledge base for metallic fuel was generated during the IFR program, in addition to knowledge base generated earlier at EBR-II with U-Fs fuels and limited experience with reprocessed fuels. This knowledge base includes ANL reports (green packs), IFR reports, EBR-II run reports, memos, PIE reports, drawings, experiments qualification reports, and publications in journals and conferences, information regarding measured properties (e.g., IFR metallic fuels handbook), and out of pile experiments. Also, measurement documents like micrographs, profilometry data, fission gas release data, and other measurements data are well documented in most cases. A lot of those documents are available in digital form as part of the IMIS database that was developed closer to the end of the IFR program. However, this database is not complete as it does not include detailed pin-by-pin data associated with the different metallic fuels experiments conducted at EBR-II. This detailed information was generated for only four experiments (X425, X430, X441, and X447). Detailed data for those experiments included pin by pin axial profiles of operating parameters for each run that the experiment was present in EBR-II, pin location in each experiment, in addition to the detailed PIE data associated with each pin. Analysis of remaining experiments depended on operating parameters for EBR-II that were generated with an older methodology that did not include pin-by-pin depletion calculations and other related details for calculating the temperatures within a subassembly and pin temperatures. Although, information from this older methodology was adequate at the time to qualify the experiments and analysis, there is a need to go back and look in detail into those experiments and associate its detailed information with the experimental observations to have thorough consistent analysis of those experiments. The newer methodology was developed close to the end of the IFR program and it was used to generate the operating parameters for each of the four experiments mentioned above this methodology depends on ANL suite of codes, REBUS/EBRFLOW/RCT/RCTP/ SUPERENERGY-II. Further effort will be needed to generate such detailed set of data for the remaining experiments and make it available to current and future analysts interested in metallic fuels. In addition, effort is needed to relate those detailed data sets to the available documentations and PIE information in an advanced database that will facilitate access to this information and connect between the experimental data and the detailed calculation for those experiments. This needs to be done by staff remaining from the IFR program that are familiar with the data and are capable of generating this comprehensive database.

A detailed description of knowledge preservation and database development for oxide fuel could not be completed in time for inclusion in this report.

IV.F Fuel Fabrication

The U.S. experience with fuel fabrication for a sodium-cooled fast reactor (SFR) comes largely from the fueling of the EBR-II and FFTF. EBR-II operated with various designs of metallic fuel in stainless steel cladding and FFTF used mixed oxide (MOX, $UPuO_2$) in stainless steel cladding. The other U.S. SFR was the Enrico FERMI reactor. The fuel design used in the FERMI reactor (U-10Mo in zirconium alloy clad) would not be used in a reactor today as the cladding material has limited use at proposed operating temperatures. The assembly design, where a Type 347 stainless steel square grid supported the fuel elements, was also unique and of little interest today.

Two recent publications (Burkes, Fielding, Porter, Crawford and Myers, 2009) (Burkes, Fielding, Porter, Meyer, and Makenas, 2009) reviewed the fabrication process development for both metallic and MOX fuel, as well as mixed carbide and mixed nitride fuel. The two journal articles provide a good bibliography of published accounts of domestic SFR fuel fabrication development.

IV.F.1Fabrication Records

Detailed specifications, procedures, and batch fabrication records would be key to facilitate a new production of these fuels without the burden of development and repeating earlier mistakes. The evolution of the specifications often reveals how lessons learned were applied to the next generation of specification.

Most of these documents resided internal to the organizations, Argonne National Laboratory for EBR-II and Hanford Engineering Development Laboratory, HEDL (aka Westinghouse Hanford Company, WHC) for FFTF and perhaps had limited distribution.

IV.F.1.a EBR-II

Fuel and assembly fabrication procedures, fuel specifications, and at least most fabrication records can be located using the Idaho National Laboratory's Engineering Document Management System (EDMS). This system documents where 'hard' copies of these materials are stored. An inventory has not been done to assess whether all relevant documents can be found there but a cursory check found that all types of this documentation were catalogued and stored. A 'word' search can be used in this system to find a listing of the location of a document being sought. Some of the documents, such as select specifications, have been scanned into the system as individual documents. If records such as these are to be made useful to future fuel development efforts they should be scanned into electronic media where they can be searched and data can be extracted as needed. These efforts have become relatively inexpensive.

Fabrication statistics were compiled for some fuel fabrication campaigns. Likewise data concerning rates of various types of rejects and other losses, returns, and pin properties (size, composition, etc.) for the most recent fuel campaigns exist that can be examined statistically to review for future process improvements. They have already been used to predict process loss for a reprocessing design. These are not currently published in the open literature.

IV.F.1.b FFTF

An inquiry was sent to individuals still working with DOE Richland area contractors who may have knowledge of fuel-related records related to FFTF fuels and experiments [Ron Omberg (PNNL) and Ron Baker (RL)]. According to Omberg, there is an active program to recover the records on FFTF fuel testing and driver fuel fabrication information, limited in rate and scope by annual funding. A summary report on FFTF codes and standards is due to be produced at the end of fiscal year 2011. The report will address both the reactor plant and the fuel and so will not focus on the fuel alone. It appears that records do exist for FFTF fuel but these individuals will not know what the extent is until funding is advanced to examine the stored records.

IV.F.2 Source of Cladding/Duct Materials and Tube/Duct Fabrication

A known 'gap' in the knowledge/experience related to fuel and assembly fabrication is the existence of an experienced and qualified supplier of materials and especially the fabrication of hardware components like tubing (cladding) and hexagonal ducts. There are no current suppliers at least if ferritic/martensitic stainless steels are the material choices.

When the Clinch River Breeder Reactor (CRBR) was going to built, FFTF was under construction, and EBR-II was a working reactor and required a steady source of hardware, there were domestic suppliers of nuclear grade stainless steel (Types 304 and 316) tubing and duct. A few of these suppliers, especially Carpenter Technologies (CarTech) and Superior Tube were funded to develop the techniques to manufacture components from what were called at the time 'advanced alloys' such as D9 (Ti-modified austenitic) and HT9 (12Cr-1Mo ferritic/martensitic, F-M). CarTech supplied HT9 tubing and duct for both FFTF and EBR-II. CarTech also drew some modified 9Cr-1Mo F-M (T91) ducts for use in EBR-II.

Recently a small scale study was undertaken to ascertain the ability of the U.S. industry to produce F-M cladding to a specification that was similar to previous FFTF and EBR-II cladding specifications. The resulting study showed that given enough scheduling time a heat of F-M material could be produced using the required double melt process of vacuum induction melting followed by vacuum arc remelting. The only real issue is the available furnaces are scheduled 18 months or more out. Although an industrial sized heat is feasible, the forming of cladding tubes is problematic. Currently there are few if any industrial applications of precision drawn F-M tubing. Tubing manufacturers can produce high quality austenitic stainless steel tubing or other alloys which are currently used in various industries, but F-M steels need specific processing parameters to produce a consistent product that can meet the precise specification of nuclear cladding. Recent efforts resulted in a low yield percentage of tubing that met both dimensional standards and surface finish standards that were similar to past cladding standards. It was assumed this was because the F-M steels must be processed differently than more common alloys. It is likely that F-M cladding could be produced; however, a significant development effort would be required to recapture the processing techniques.

There have been very recent inquiries by commercial nuclear interests into providing core loadings of HT9 or T91 duct and cladding. Kobe Steel, Japan, has shown interest and competency with providing similar materials of nuclear grade. The lack of domestic, qualified suppliers is a 'gap' in domestic technology that should be closed at first onset of a new mission to build a fast reactor.

IV.F.3 Material and Fabrication Development – The National Cladding and Duct (NCD) Development Program

The Fast Breeder Reactor (FBR) development program contained a large effort to develop materials and design for use as fuel cladding and assembly ducts. This was the National Cladding and Duct (NCD) Program. The Hanford Engineering Development Laboratory (HEDL, aka Westinghouse Hanford Company [WHC], GE Nuclear, Westinghouse – Advanced Reactors Division (WARD), Argonne National Laboratory, Oak Ridge National Laboratory and the Naval Research Laboratory were the primary participants in this work and held regular inter-laboratory meetings to review data and develop design codes to represent various material properties and effects. A large amount of what is known about the performance of these stainless steel materials in an irradiation environment was produced in this program. Unfortunately much of the work was never published because at the time the work was considered Applied Technology, a categorization designed to keep the information within a very limited distribution. At the end of the program the materials were studied for application to fusion reactors. The new funding organization allowed this later work to be published openly and does not require an information recovery process.

The older, and larger, body of this work was largely documented in a Quarterly Report (HEDL TC-160-XX). It needs to be recovered to allow access and in a way so it may be word-searched. The quarterlies consisted of a series of short papers on specific subjects such as, "ANALYSIS OF RESULTS FROM THE SECOND INTERIM EXAMINATION OF THE ADVANCED ALLOY CREEP IN BENDING EXPERIMENT" There is often no index or listing of the articles contained within a report but they are arranged by general subject areas, such as creep, swelling, microstructure, simulation (ion irradiation, etc.), mechanical properties, coolant compatibility, fabrication, etc., usually five or six at a time that comprised the NCD program. However, these subject areas changed several times during the life of the NCD Program. It is therefore very difficult to locate the information for which you are searching without a word-searchable database, and currently that does not exist. Since much of the information in these 'Applied Technology' reports has never been published and were not otherwise available, we are destined to repeat the work if such a database is not created and advertised to the new generation of researchers.

The National Cladding and Duct (NCD) Quarterly Reports (HEDL TC-160-xx) contain some of the only documented accounts of developing the fabrication methods used to make fuel cladding and assembly hardware from alloys such as HT9. For a time the NCD quarterlies did have a chapter, "Group D, 'Fabrication and Development'" and this is where some of this information can be found. A scanned searchable database is needed to make this important information useful to current-day researchers and development engineers, and to help to qualify hardware vendors.

IV.H Transient Behavior

Behavior of advanced fuels must be acceptable under a wide range of normal and off-normal environments and conditions that can potentially arise in sodium cooled fast reactors (SFRs) (Wright, Dutt and Harrision, 1990). Specifically, qualification of fuels and the approval of reactor designs include understanding and reliable prediction of the transient behavior of fuels and cores under the full range of anticipated and postulated conditions through cladding breach and beyond. Thus, in addition to having proven, excellent performance, fuels must be shown to have acceptable behavior under off-normal conditions as well as design-basis accident conditions and beyond -- as needed for approval of lead test assemblies and in plant licensing.

Off-normal conditions will arise from local defects and/or plant transients. These may potentially result in local fuel damage, cladding failures, and/or extensive fuel damage. Outcomes will depend in large part upon the action of on-line diagnostics, operator actions, and automatic plant protection systems. Outcomes will also depend on key properties of fuel, cladding, and core structural materials at elevated temperature -- specifically, the interactions and compatibility of those materials with each other and with the sodium coolant. Increasing severity of off-normal conditions to which fuel might be subjected may result in an expansion of the types and ranges of phenomena that characterize the response and the damage that may result.

Historically, fuel behavior studies for oxide and metallic fuels have extended far beyond characterization of fuel and cladding behavior under normal conditions. In addition, models and codes were developed to specifically address transients for conditions short of cladding breach as well as those associated with severe accidents. For SFRs, the models and codes needed for such analyses typically involve a combination of thermal-hydraulics, mechanics, neutronics, and materials behavior. Furthermore, fundamental differences in the transient behavior between oxide fuels and metallic fuels resulting from local faults or whole-core accident initiators have required separate modeling and validation bases.

Because early (1960s) metallic fuel designs were limited to much lower burnup than were oxide fuel designs, transient behavior of oxide fuels was studied far more extensively than that of metallic fuels for the next two decades. However, during the 1970s and 1980s, interest in metallic fuels increased due to advances in understanding of fuel behavior, improved irradiation performance, attractive fabrication and reprocessing features, and favorable transient behavior characteristics of that fuel form. To date, both up-to-date oxide and metal fuel designs have performed well under normal conditions at least for burnups up to about 10-12 at%, and both fuel types have reached burnups as high as 19-20 at%. Cladding failure thresholds (expressed in terms of power-over-flow ratios) are similar for both oxide and metal fuel types. Nevertheless, the demonstrated response of both fuel forms to severe accidents has prompted work to further improve existing designs.

With oxide fuels, there is motivation to reduce the probability of molten fuel being released from cladding into coolant channels causing energetic coolant expulsion from the core and complete coolant channel blockage by freezing fuel and/or cladding, resulting in a severely-disrupted, heat-generating, uncooled, metastable core configuration which might become re-critical. With metallic fuels, iron-based cladding and structural materials are vulnerable to the formation of low-melting-point compositions with uranium- (and plutonium-) fuel materials. While formation of such molten phases is a principal cause of cladding failure in metallic fuels, there appears to be little tendency to form coolant flow blockages. The gap between current knowledge and what may be needed for licensing, either for metal or oxide fueled cores, is associated with future designs capable of reaching high burnups (20 at% and above) along with improved safety performance.

The transient behavior of SFR oxide fuels of current designs through at least the mid-1980s has been extensively investigated and is generally well predictable with fuel behavior and whole-core-accident codes (on a fuel macro-scale, up to medium burnups, and with significant reliance on empirical correlations). Extending those codes to describe improvements in fuel pin and subassembly designs capable of reaching fuel burnups of 20 at% and above along with improved safety performance will require additional validation and model development. Examples of oxide design improvements under study include annular fuel with annular axial blanket/reflector to facilitate pre-failure molten fuel axial dispersal, longer-life cladding materials such as oxide-dispersion-strengthened steel, and sub-assembly designs to facilitate axial flow of molten core materials out of the core during postulated subassembly or whole-core melt accidents.

While experience with metallic fuel has been considerable, the empirical and analytical knowledge base for metallic fuels is considerably smaller than for oxide fuels. In particular, transient testing of modern metallic fuels has not been nearly as extensive as was performed for oxide. Correspondingly, performance and accident codes for metallic-fuel are less well developed. Proposed fuel design improvements in metallic fuel to prevent fuel-cladding chemical interaction (such as fuel additives or cladding liners or coatings) will need to be investigated regarding their efficacy during normal operation and influence upon the fuel transient response.

Overall, little is known about the steady-state performance or transient response characteristics of metal or oxide fuels and cores either:

1. At high burnups (> 12 at%),
2. With recycle fuels, or
3. With fuels having high minor actinide content.

Even less is known about the steady-state and transient response of future fuel design improvements to reach burnups well above 20 at%, such as fuel pins having lower fuel smear densities and advanced claddings.

Evaluating advanced fuels and core designs regarding such key issues as margin to fuel melting, margin to cladding failure, and acceptable response beyond cladding failure will need to be a continuing activity. A high importance level of such issues coupled with a minimal existing knowledge base to resolve those issues leads to a conclusion that the closing the knowledge gaps described above is of high priority [see "Experimental Facilities for Sodium Fast Reactor Safety Studies," 2011 OECD report NEA/CSNI/R(2010)12] and requires the availability of suitable transient testing capabilities.

IV.I Structural Materials

Due to the unavailability of the proper expertise during the panel meeting, regulatory gaps in structural materials were not directly considered by the panel. Instead, this report leveraged a number of previous studies which considered the current state of SFR structural materials. This section summarizes the findings of these studies (Chopra and Natesan, 2007) (Natesan et al, 2008).

An objective of these reports was to evaluate the licensing and design implications of the ASME code qualification on an SFR (Natesan et al., 2008). It was noted that Clinch River Breeder Reactor (CRBR) and the Power Reactor Innovative Small Module (PRISM) faced regulatory questions concerning compliance with the elevated temperature structural integrity criteria (ASME Code Section III Subsection NH). It should be noted that Subsection NH has not been approved by the NRC and a version of Subsection NH will need to be adopted by the NRC before an SFR can be licensed in the US. Currently, only 5 alloys are included in Subsection NH including: 304SS, 316SS, 2.25Cr-1Mo, Alloy800H, and Mod. 9Cr-1Mo (grade 91).

Thirteen major gaps were identified:

- Lack of materials property allowable data/curves for 60 year design life
- Lack of validated weldment design methodology
- Lack of reliable creep-fatigue design rules
- Lack of hold time creep-fatigue data
- Improved mechanistically based creep-fatigue life predictive tools are needed for reliable extrapolation of short term data to 60 year life

- Lack of understanding/validation of notch weakening effects
- Methodology for analyzing Type IV cracking in 9Cr-1Mo weldment
- Lack of inelastic design procedures for piping
- Lack of validated thermal striping materials and design methodology
- Material degradation under irradiation
- Materials degradation under thermal aging
- Materials degradation in sodium environment
- Degradation under sodium-water reaction

Tables 16 and 17 list the materials historically used in SFRs and their associated Subsection NH limits.

Table 16. Materials Used in Past Sodium-Cooled Reactors

Country	Reactor	Vessel	Intermediate Heat Exchanger (IHX)	Steam Generator Evaporator	Steam Generator Superheater
USA	Fermi	304	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
	EBR-II	304L	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
	FFTF	304	316	a	a
	CRBR	304	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
UK	DFR	316	316	321	321
	PFR	321	321	Fe-2¼Cr-1Mo	316H
Russia	BOR-60	304	304	Fe-2¼Cr-1Mo (Alloy 800)	Fe-2¼Cr-1Mo
	BN-350	304	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
	BN-600	304	304	Fe-2¼Cr-1Mo	304
Germany	SNR-300	304	Fe-2¼Cr-1Mo-1Nb	Fe-2¼Cr-1Mo-1Nb	Fe-2¼Cr-1Mo-1Nb
France	Rapsodie	316L	316	a	a
	Phenix	316L	316	Fe-2¼Cr-1Mo	321
	SuperPhenix	316	316	Alloy 800 tubes and 304, 316L shell	b
Japan	Joyo	304	304	a	a
	Monju	304	304	Fe-2¼Cr-1Mo	304

[a]Sodium to air heat exchanger.
[b]Evaporator and superheater are combined in a single unit.

Table 17. Materials included in Subsection NH allowable

Material	Temperature (°C) Primary stress limits[a]	Temperature (°C) Fatigue
304	816	704
316	816	704
2.25Cr-1Mo	593[b]	593
Mod.9Cr-1Mo	649	538
800H	760[b]	760

[a]Allowable stresses extend to 300,000 h (34 years).
[b]Temperatures up to 649°C allowed for no more than 1000 h.

Tables 11-15 summarizes potential systems and components of an SFR and identifies gaps in fabrication and performance.

Table 11 indicates that little additional research efforts are needed to develop Reactor System Structural Components. All ranked structures and components are satisfactory in fabrication, degree of knowledge and sufficiency of the structure or component to complete the desired mission. The only gap identified was the design of the rotatable plug for the reactor vessel head.

Tables 12 and 13 indicate that additional work is needed for all primary and secondary structures and components in the heat transport system. The Electromagnetic pump was identified as a major gap.

Table 14 indicates that a large research effort will be needed if the Supercritical CO_2 (S-CO_2) Brayton cycle is to be incorporated into an SFR design. Most components, excluding potentially the Recuperator, need to select potential materials, develop fabrication capacity and improve the experimental database. Much of this work is now being conducted at the S-CO_2 test loop at Sandia National Laboratories.

Table 15 indicates that a Rankine power conversion cycle could be incorporated into an SFR design with minimal additional work needed to improve the technology status of the steam generator shell, steam generator tubing, and hot leg steam piping.

V. Identification and Discussion of Significant Gaps

The current state of knowledge of SFR fuel and structural material performance is sufficient for designing and licensing an SFR today within the envelope of a conservative data base. The boundaries of a conservative data base would be a fuel burnup of 10 at% or less, either metallic or oxide fuel, a peak cladding temperature of 600^{o}C or less, a peak dpa of 100 or less, and with fuel that has not been reprocessed. Both the steady-state and off-normal irradiation data base would be sufficient to support such a design. The only qualifications to the above statement are the following: The existing data must be retrievable and in a form, from a QA standpoint, that is acceptable to the licensing body. Fabrication experience for fuel, cladding, and ducts must also be retrieved to provide assurance that the core materials could be replicated such that the existing data base is applicable. It must be appreciated that few, if any, vendors of these materials exist. Thus for fuel from zero to moderate burnup, two gaps exist:

1. An effort should be made to inventory the existing data base, collect the hard copy information and store it in approved storage locations, and transfer this information to an electronic data base that can be readily queried.
2. Exactly the same effort should be carried out for the fabrication information.

A reactor designed fuel burnup up to 20 at% will have a database weakens substantially for both metallic and oxide fuel. The number of fuel pins taken to 20 at% is limited and these pins were not taken to high burnup without reconstitution. Thus, there is no whole assembly experience or whole core experience at high burnup. Without the availability of a test reactor such a design to high burnup could not be licensed. Thus, the major gaps for fuel irradiated beyond 10 at% are the following:

1. A need for irradiation of a significant number of prototypic assemblies to high burnup in the steady state conditions.
2. Subject a number of high burn pins to off-normal (TREAT) tests.

SFRs have been viewed as means to fission the minor actinides, americium, neptunium, and curium that arise from the reprocessing of LWR fuel in order to reduce the heat load and radio-toxicity of a spent fuel repository. As shown on Table 3 the technological data base is weak for either oxide or metal fuel that contains substantial quantities of minor actinides. Experiments are underway in the ATR reactor to study the performance of metal and oxides fuel that contain additions of minor actinides. However, the fuel capsules are small and the neutron energy spectrum does not duplicate that of a fast reactor. The following gap exists for fuel with additions of minor actinides:

- Irradiation data gained from ATR must eventually be augmented with the irradiation of full-size capsules in a SFR test reactor or modeled to the extent that the results from the small ATR capsules can be convincingly extrapolated to full size fuel pins

It is unlikely that SFR fuel would be reprocessed with PUREX, which has minimal carry-over of fission products to the reprocessed fuel. Pyro-processing or UREX have the potential for substantial carry-over of fission products. As shown in Table 2 the data base is weak for the performance of either metal or oxide fuel that contains a substantial quantity of carried over fission products. For oxide fuel fabrication may be problematic, while for metal fuel the migration of lanthanide fission products to the fuel cladding interface may result in low melting compounds. Experiments are underway in ATR to aid in the resolution of these issues. Thus the gap identified in this area is identical to that identified above for fuel that contains additions of minor actinides.

The last U.S. variation of 316 stainless steel, that being cold-worked with titanium and other alloy additions, designated as D9, is suitable for both oxide and metallic fuel cladding and ducts up to modest burnup levels and dpa less than 100. Vendors for this steel are readily available. The only identified gap was that more information is needed relative to fuel-cladding chemical interaction for reprocessed fuel with fission product carry-over, particularly the issue of lanthanide migration to the fuel-cladding interface in metallic fuel.

The ferritic/martensitic alloys have the potential to solve the irradiation enhanced swelling issue for both cladding and ducts up to at least 150dpa and perhaps 200dpa[5,6]. However, the majority of the high dose data originates from a duct that operated at a relatively low temperature compared to fuel cladding temperatures. Thus the following gaps exist for both HT-9 and for the advanced cladding T91 (9Cr1Mo):

1. High dose-high temperature swelling data do not exist for HT-9 or T91. Any data that exist or will be generated will originate from foreign SFRs.
2. Recent attempts to obtain a small heat of HT-9 revealed that there are no vendors readily available to produce reactor grade material.

Several gaps were identified in the discussion of fuel performance codes.

1. Virtually all the gaps were related to the fact that there has been little attention given to fuel performance code development for the last two decades. Most of the code routines are empirically based as opposed to mechanistically based and thus are useful primarily for interpolation when adequately validated with existing data.
2. In addition, relatively few people are adept in exercising the codes with documentation less than adequate for the training of new users.

In the area of structural materials it was noted that the panel borrowed from the results of previous gap analyses. It was generally concluded that should an SFR be designed in the near future, based on a Rankine cycle that the technology base was likely adequate to license the reactor, provided that the burnup was limited to 10 at%. The only exceptions were the design of the rotating plug and the lack of performance data should a large primary electro-mechanical pump be part of the design. The overall materials technological base for the Brayton cycle would require a significant research effort, though this cycle offers many advantages to more traditional power cycles.

Two overarching gaps were apparent throughout the gap analysis discussions. These were:

1. For most of the identified gaps a test SFR such as EBR-II or FFTF is required to enhance the existing knowledge base.

2. The state of the existing knowledge base is uncertain. Operating information, fuel performance data, and fabrication experience exists in a number of locations. Some exists on electronic media, which may or may not be queried easily, some on hard copy reports that are stored in substandard locations, and some may be lost.

It is extremely important to preserve the existing data base because without EBR-II, FFTF, and TREAT the information cannot be duplicated. Even in the event that such facilities become available in the future, duplication of these irradiations would be expensive and time consuming.

VI. Conclusions

The main conclusion reached by the panel was that an SFR could be designed and licensed based upon the technology base developed from the successful operation of EBR-II and FFTF. However, the design would be constrained within the limitations of the technology base.

From a fuels and materials perspective these limitations are the following:

1. For either oxide or metal fuel a maximum burnup of 10 at%.
2. A peak cladding temperature of 600^{o}C.
3. The use of D9 stainless steel cladding and duct material.
4. A peak irradaition exposure of 100 dpa on the cladding and duct.
5. The use of fresh fuel (fuel with neither additions of minor actinides or fission product carry-over from reprocessing)
6. Limited-load following operation for oxide fuel

For burnups greater than 10 at% for either fuel type, the data are limited up to 20 at% and nonexistent beyond 20 at%. For cladding temperatures above 600^{o}C the strength of D9 diminishes rapidly, and irradiation data for higher strength alloys is sparse. The capability to manufacture components from austenitic steels such as D9 exists, but for advanced alloys the industrial base needs to be developed.

Although neutron exposures for advanced alloys such as HT-9 exist up to 150 dpa, the data are meager. Research is on-going in the national labs to study the effects of minor actinide additions and fission product carry-over for both oxide and metal fuels. However, the work is only partially completed.

Metal fuel has been shown to be robust when subjected to load-following conditions, whereas these data are lacking for SFR oxide fuel.

A primary concern of the panel was the status of the existing technology base for SFR fuels and materials. If a serious attempt were made to license a SFR, could the information be retrieved in a credible form to be used for licensing? Many of the scientists and engineers who were involved in the data generation are no longer in the workforce, and much of the information resides in a number of locations in a variety of formats such as electronic media, internal reports, and publications. It would be prudent to form a task group to assess the state of the technology base and provide recommendations for long-term preservation of the information.

Although SFR fuels and materials have the potential to operate well beyond the limitations expressed above, it is difficult to extend the technology base without the availability of test facilities such as EBR-II, FFTF, and TREAT. Many of the gaps that were identified such as higher burnup, higher dpa, etc, depend on the availability of test facilities. Without domestic facilities and with limited access to diminishing foreign capability, SFR designs will be limited to the existing base that must be preserved for future SFR designs.

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