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PRELIMINARY EVALUATION OF LICENSING ISSUES ASSOCIATED WITH U.S.-SITED CANDU-PHW NUCLEAR POWER PLANTS

by

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PREFACE

An evaluation of the licensing aspects of a specific type of nuclear power plant covers a wide area of expertise and cuts across many disciplines. A detailed study of this nature requires a considerable amount of time and effort. The present study was limited in both time and effort, the primary objective being a preliminary identification and evaluation of those licensing aspects that would be associated with the siting in the US of CANDU-PHW Power Plants, in view of the current US-NRC requirements.

Specifically, this study is limited in that no independent safety analyses were performed and that no independent evaluation was made regarding the data base supporting the safety analyses. The conclusions drawn in this report are therefore, to the extent that they depend on safety analysis, based solely on the results of analyses performed by AECL, the designer of CANDU-PHW nuclear power plants.

The above qualifying statement is in no way to imply any doubt regarding either the quality or the depth of the safety analyses performed by AECL; it is solely made to clarify one of the limitations of the present study. A complete licensing assessment would have to include a more detailed evaluation of both the safety analysis and the supporting data base.

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PRELIMINARY EVALUATION OF LICENSING ISSUES ASSOCIATED WITH U.S.-SITED CANDU-PHW NUCLEAR POWER PLANTS

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ABSTRACT

The report briefly describes the principal safety-related characteristics of current CANDU-PHW power plants, and makes a distinction between those characteristics which are intrinsic to the CANDU-PHW system and those that are not.

An outline is given of the main features of the Canadian safety and licensing approach. Differences between the U.S. and Canadian approach to safety and licensing are discussed. Some of the main results of the safety analyses, routinely performed for CANDU-PHW reactors, are presented.

U.S.-NRC General Design Criteria are evaluated as regards their applicability to CANDU-PHW reactors; vice-versa the CANDU-PHW reactor is evaluated with respect to its conformance to the U.S.-NRC General Design Criteria.

A number of design modifications are proposed to be incorporated into the CANDU-PHW reactor in order to facilitate its introduction into the U.S. These modifications, which consist of both deletions and extensions to safety-related systems in current CANDU-PHW reactors, represent a trade-off which is proposed solely for the purpose of maintaining consistency within the current U.S. licensing environment, in particular with respect to nuclear reactor types already being licensed in the U.S.; these modifications are <u>not</u> proposed out of a need to improve the safety of the current CANDU-PHW reactor design.

A number of issues are identified which still require resolution. Most of these issues relate to design areas not yet covered by the ASME code. This latter situation, which is primarily attributable to the fact that there has not existed in the U.S. any strong need for ASME Code development in areas of interest to pressure tube reactors, does not in any way bring in question the validity of the design solutions followed for the CANDU-PHW reactor.

I. Introduction

The Canadian approach to nuclear reactor safety, while having been influenced by developments elsewhere, has to a large extent been developed independently.[1-31] This fact, combined with the fact that the CANDU-PHW (CANDU-Pressurized Heavy Water) reactor has a number of intrinsic safety-related characteristics differing substantially from those of other commercial nuclear power reactor types, has led to the development of Canadian licensing criteria and safety design bases which are in some areas different from those developed, e.g., in the US for light water reactors (LWRs).

The primary issue in evaluating CANDU licensability in the US is therefore not whether CANDU reactors meet adequate safety standards, but rather how much effort is required to introduce the CANDU technology into the U.S. regulatory environment, as determined by current U.S.-NRC regulatory criteria, guides, procedures, practices, and standards.

The evaluation of the licensability of US-sited CANDU reactors can be approached in a number of ways. One way would be to follow a purely probabilistic approach of the WASH-1400 type, comparing over the entire spectrum of postulated accident sequences the overall probability of each type of accident sequence for CANDU reactors and LWRs. This approach, which would appear to have considerable merit, has certain difficulties associated with its application. One important difficulty is that current US licensing procedures are up to now only to a relatively small degree based on probabilistic considerations.

The approach followed in this report, which seems to parallel to some extent that applied up to now for the liquid metal cooled fast breeder reactor (LMFBR) in the US, is that of <u>equivalency of safety</u>, which for the purposes of this report is understood to mean that a US-sited CANDU reactor is to have, in all areas, safety levels (margins) that are equal to, or higher than, those of US-sited LWRs, making proper allowances for the differences between the safety-related intrinsic characteristics of CANDU reactors and LWRs.

It should be recognized that US licensing criteria and regulations (including the General Design Criteria, presented in Title 10 of the Code of Federal Regulations, Part 50, Appendix A - 10 CFR 50, Appendix A) have, to a very large extent, been evolved around the present generation of LWRs. The Code of Federal Regulations recognizes this fact, stating: "These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units." In view of this, some of these criteria and regulations are not (or only partially) applicable to CANDU reactors, while others should be interpreted as to their intent rather than their specific wording. In some cases a re-wording of some of the current regulations and criteria may be desirable so as to be able to cover both LWRs and CANDU reactors.

Design characteristics, which are typical for the CANDU reactors and different from current LWRs, require special attention in that some of them may not have been addressed up to now in the U.S. regulatory process or in current US standards (ASME, etc.). An example is the use of Zr-Nb alloy as part of the primary coolant pressure boundary (such as is the case for the CANDU in-core pressure tubes), which did not have to be addressed for LWRs.

In evaluating the safety aspects of CANDU reactors with respect to licensability in the US, it is desirable to clearly distinguish design characteristics that are <u>intrinsic</u> to the Nuclear Steam Supply System (NSSS) from those that are <u>non-intrinsic</u> (i.e., pertaining to subsystems of a more peripheral nature), and that could therefore be modified relatively easily without changing any of the principal characteristics of the NSSS.[29] In the latter category are the design characteristics associated with such systems as containment, most parts of the control and plant protection, auxiliary feed water supply, most engineered safeguards, etc.

As a matter of interest, it should be noted that, in Canada in the field of nuclear power, separation between the Regulatory Function (performed by the Atomic Energy Control Board - AECB) and the Development Function (performed by Atomic Energy of Canada Limited - AECL) has existed since the early days. The Canadian Atomic Energy Control Act of 1946 established the Atomic Energy Control Board to issue regulations governing all aspects of the development and application of nuclear energy. The development functions in the nuclear field was first performed by the National Research Council, and subsequently transferred (since 1945) to Atomic Energy of Canada Limited.

II. Brief Description of CANDU-PHW Reactors

CANDU-PHW reactors belong to the family of heavy-water-moderated pressure-tube reactors, of which Table II.1 gives a non-exhaustive list of types, developed or under development for commercial application. Among the main characteristics of the CANDU-PHW should be named (1) use of pressure tubes in the core region, (2) use of D_2O as moderator (cold), (3) use of pressurized ($\sim$ 1400 psia) D_2O as coolant, (4) use of natural uranium as fuel (up to now; could be slightly enriched, if desired for higher burnup), and (5) use of an on-load refuelling scheme.

Table II.2 gives the evolution of CANDU-PHW reactors as examplified by some of the important plant parameters.

Figures II.1 through II.20 show some of the main design characteristics of the CANDU nuclear power plant. For a detailed description of the CANDU reactor reference is made to the literature. Some observations concerning the figures are as follows:

Figure II.1 gives a simplified flow diagram showing the main characteristics of the PHTS. It is noted that a single loop consists of a "figure-of-eight" configuration comprising as main components: two pumps, two steam generators, four headers, and a large number of feeder lines and power channels. Figure II.2 gives the PHTS layout for a two-loop plant, showing the two "figure-of-eight" configurations. Figures II.3, II.4, and II.5 give layouts of the core, the calandria, and the PHTS.

Figure II.6 gives a sketch of the 37 element fuel bundle used in CANDU-PHW reactors.

Figures II.7, II.8, and II.9 give cutaway views of the reactor building of, respectively, a single-unit standard plant, a multi-unit station with vacuum containment and with steam generators and PHTS pumps inside the containment, and a multi-unit station with vacuum containment with steam generators and PHTS pumps partially outside the containment.

Figures II.10, II.11, and II.12, and II.13 give layouts of the calandria and its concrete vault.

Figures II.14 gives a layout of the various reactivity control devices.

Figures II.15 and II.16 show the main features of the fueling machine and the fuel handling system.

Figure II.17 shows the PHTS and its auxiliary circuits.

Figures II.18 and II.19 show the layouts of, respectively, the Moderator Cooling System, and the Shutdown Cooling System.

Figure II.20 gives a schematic representation of a single-unit containment system as applied for 600 MWe standard plants.

The CANDU-PHW reactor has the attractive feature of permitting, during normal operation, location and removal of failed fuel elements that have become a source of radioactive contamination of the PHTS. This feature greatly assists in keeping a low level of radioactivity in the PHTS.

Table II.1 Heavy-Water-Moderated Pressure-Tube Reactors

Name	Coolant	Flow-Regime	Fuel Enrichment	Void-Reactivity Effect	Developing Country of Organization
CANDU-PHW[a]	D_2O	Single-Phase	Natural	Slightly Positive	Canada
CANDU-BLW[b]	H_2O	Two-Phase $T_i < T_{sat}$	Natural	Positive	Canada
CANDU-OCR[c]	Organic	Single-Phase	Natural	Positive	Canada
SGHWR[d]	H_2O	Two-Phase $T_i < T_{sat}$	Slightly Enriched	~Zero (undermoderated)	UK
CIRENE[e]	H_2O	Single-Phase/ Two-Phase $T_i > T_{sat}$	Natural or Slightly Enriched	Positive	Italy
EL-4[f]	CO_2	Single-Phase	Natural	-	France
FUGEN	H_2O	Two-Phase $T_i < T_{sat}$	Slightly Positive	Zero or Slightly Positive	Japan
ORGEL[g]	Organic	Single-Phase	Natural or Slightly Enriched	Positive	Euratom

[a]CANDU-PHW: CANDU-Pressurized Heavy Water.

[b]CANDU-BLW: CANDU-Boiling Light Water.

[c]CANDU-OCR: CANDU-Organic Cooled Reactor.

[d]SGHWR: Steam Generating Heavy Water Reactor.

[e]CIRENE: CISE Reattore a Nebbia.

[f]EL-4: Eau Lourde-4.

[g]ORGEL: Organique-Eau Lourde (discontinued program).

TABLE II-2 Evolution of CANDU-PHW Power Reactors

	DOUGLAS POINT	PICKERING A	BRUCE A	GENTILLY 2*
Net Output (MWe)	208	514 x 4	745 x 4	600
Number of Channels	306	390	480	380
Core Length (cm)	500	594	594	594
Fuel Inventory (Mg U)	41.5	92.3	114	95.8
Burn-up (MWd/Mg U)	8400	8000	9600	7500
D_2O inventory (Mg)	179.5	403.69	568.1	467
Inlet Temperature (°C)	249	249	252°C inner region 264°C outer region	267
Outlet Temperature (°C)	293	293	299	312
Number of Pumps	10	16 (12 active)	4	4
Number of Boilers	8	12	8	4
Turbine				
Steam Temperature (°C)	250 at throttle	250 at boiler	253	258
Throttle Pressure (MPa)	4.05	4.02	4.13	4.54

*typical of 600 MW(e) design

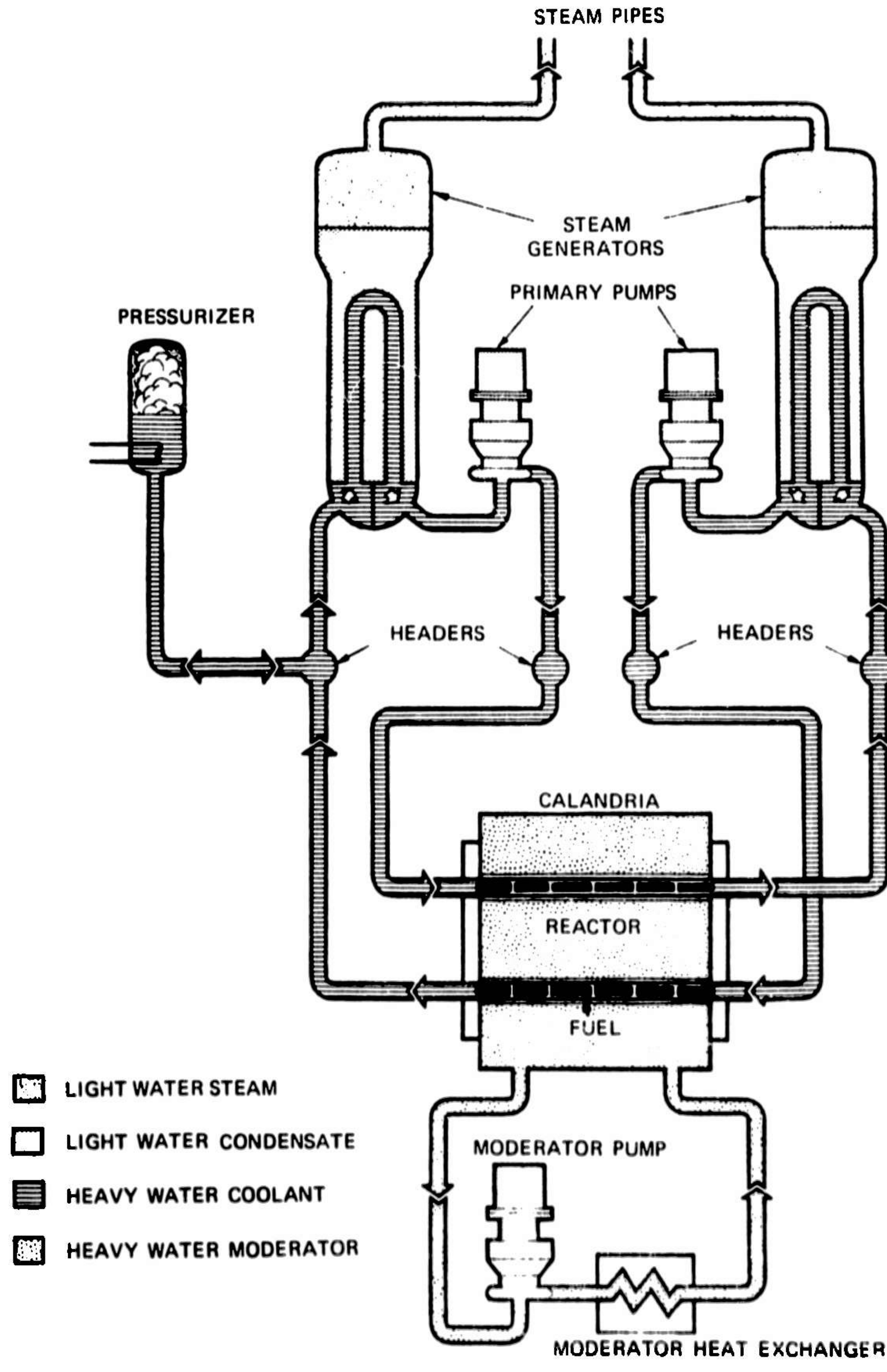


Fig. II.1 CANDU Reactor Simplified Flow Diagram

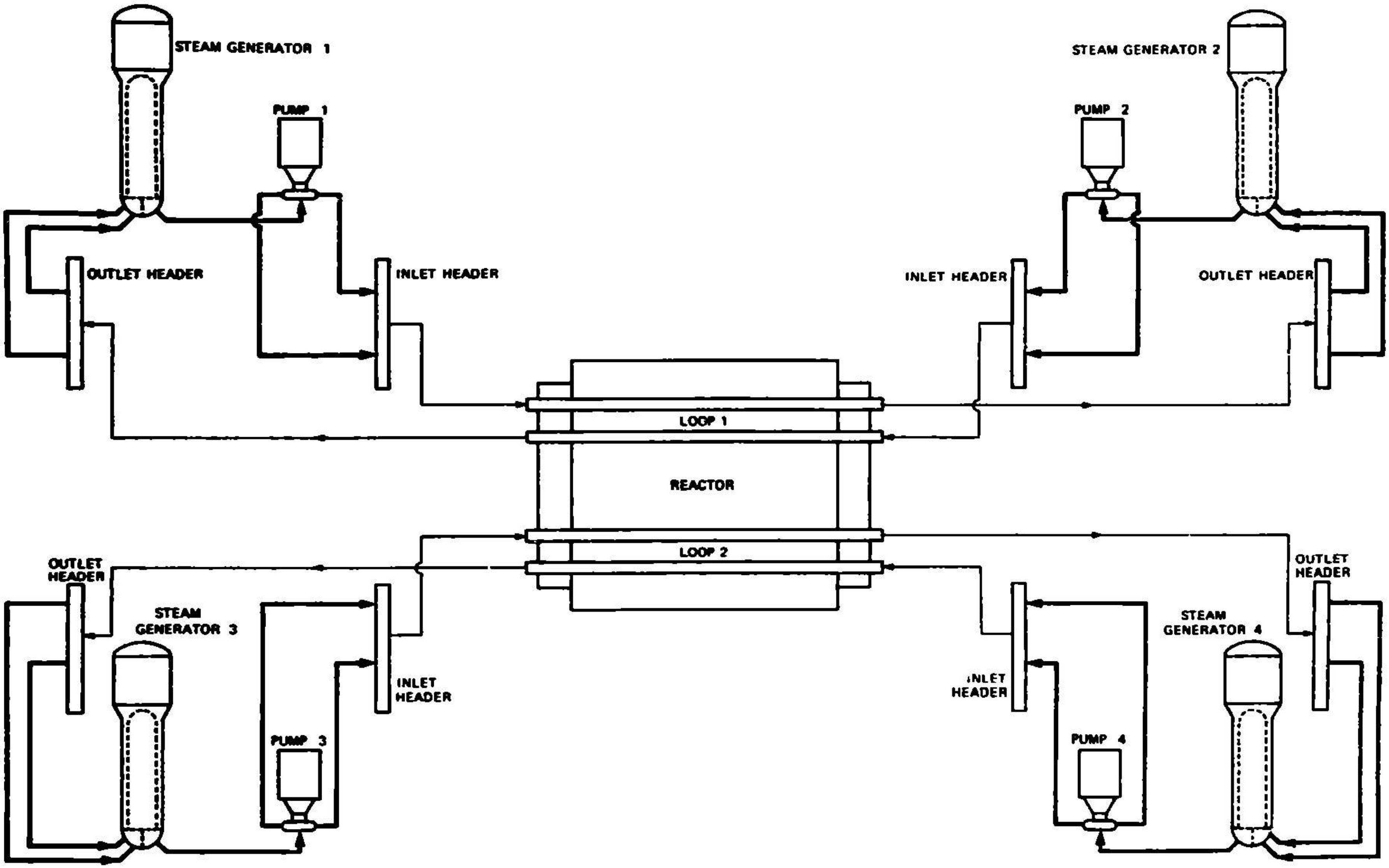
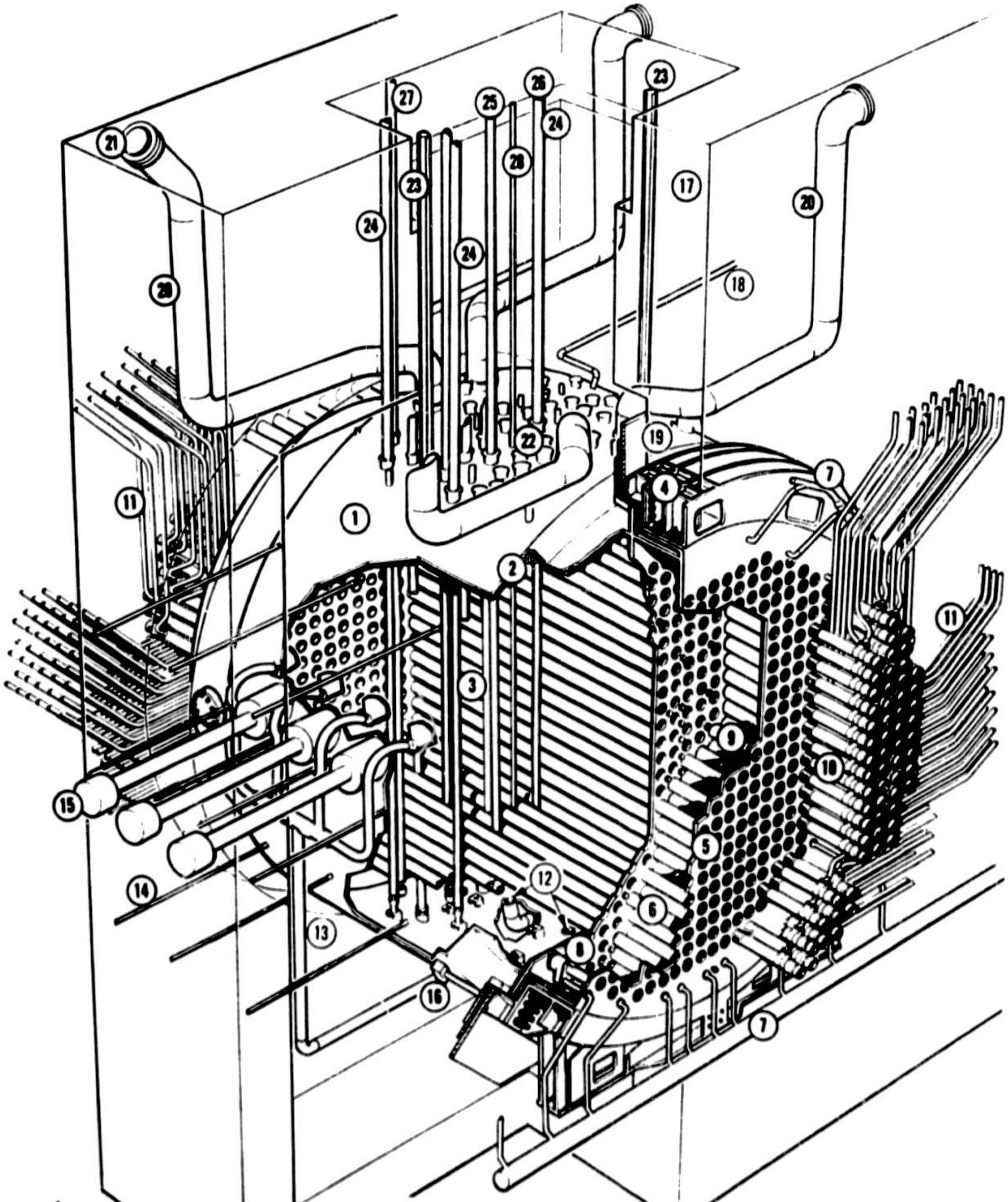


Fig. II.2 Primary Heat Transport System - Main Circuit Flowsheet



1 CALANDRIA
2 CALANDRIA SHELL
3 CALANDRIA TUBES
4 EMBEDMENT RING
5 FUELLING TUBESHEET
6 END SHIELD LATTICE TUBES
7 END SHIELD COOLING PIPES
8 INLET OUTLET STRAINER
9 STEEL BALL SHIELDING
10 END FITTINGS
11 FEEDER PIPES
12 MODERATOR OUTLET
13 MODERATOR INLET
14 FLUX DETECTOR AND LIQUID INJECTION NOZZLE
15 ION CHAMBER
16 EARTHQUAKE RESTRAINT
17 VAULT WALL
18 MODERATOR EXPANSION TO HEAD TANK
19 CURTAIN SHIELDING SLABS
20 PRESSURE RELIEF PIPES
21 PRESSURE RELIEF DISC
22 REACTIVITY CONTROL ROD NOZZLES
23 VIEWING PORT
24 SHUTOFF ROD
25 ADJUSTER ROD
26 CONTROL ABSORBER ROD
27 ZONE CONTROL ROD
28 VERTICAL FLUX DETECTOR

Fig. II.3 Reactor Assembly

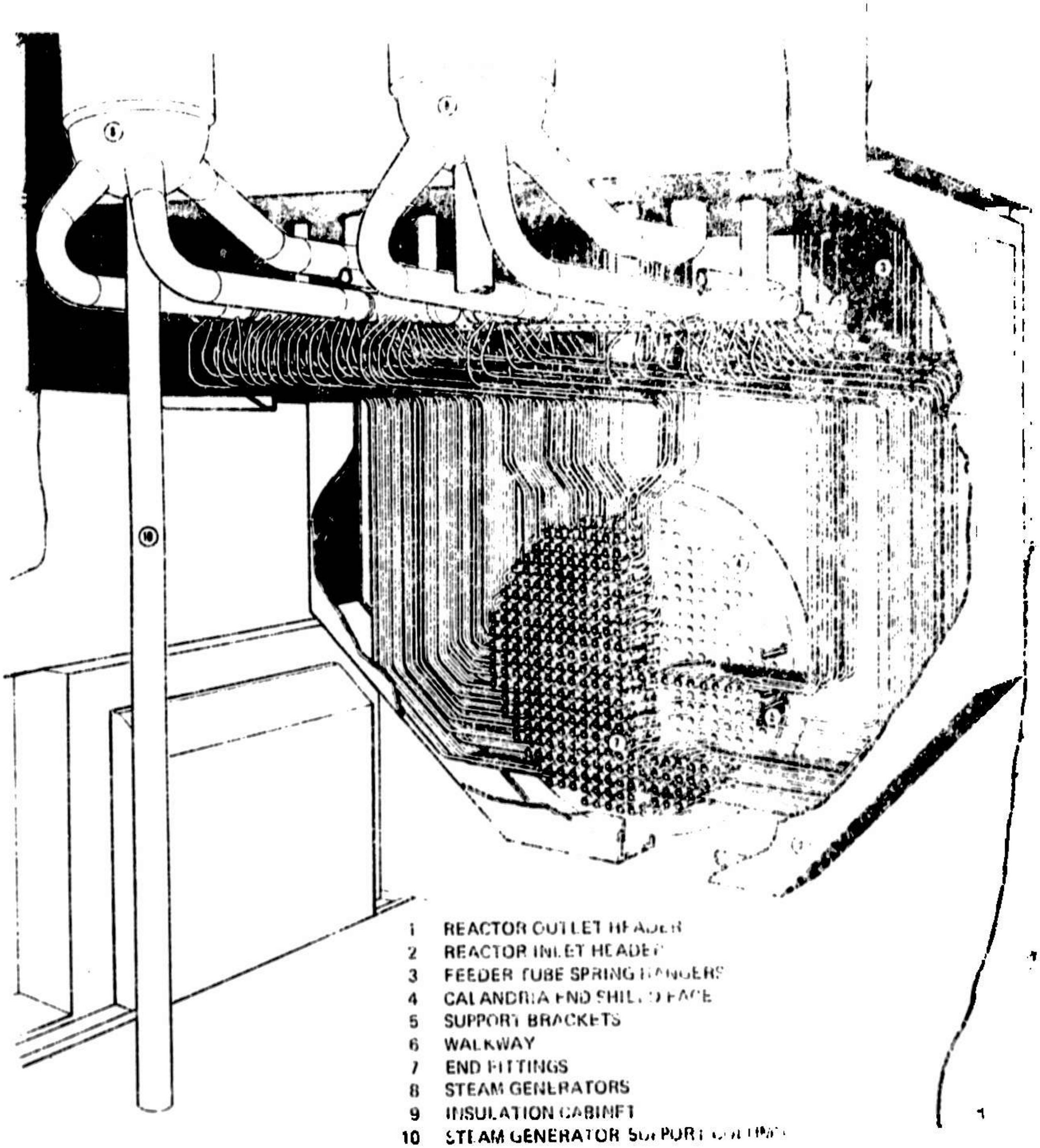


Fig. II.4 Primary Heat Transport System - Typical Feeder and Header Arrangement

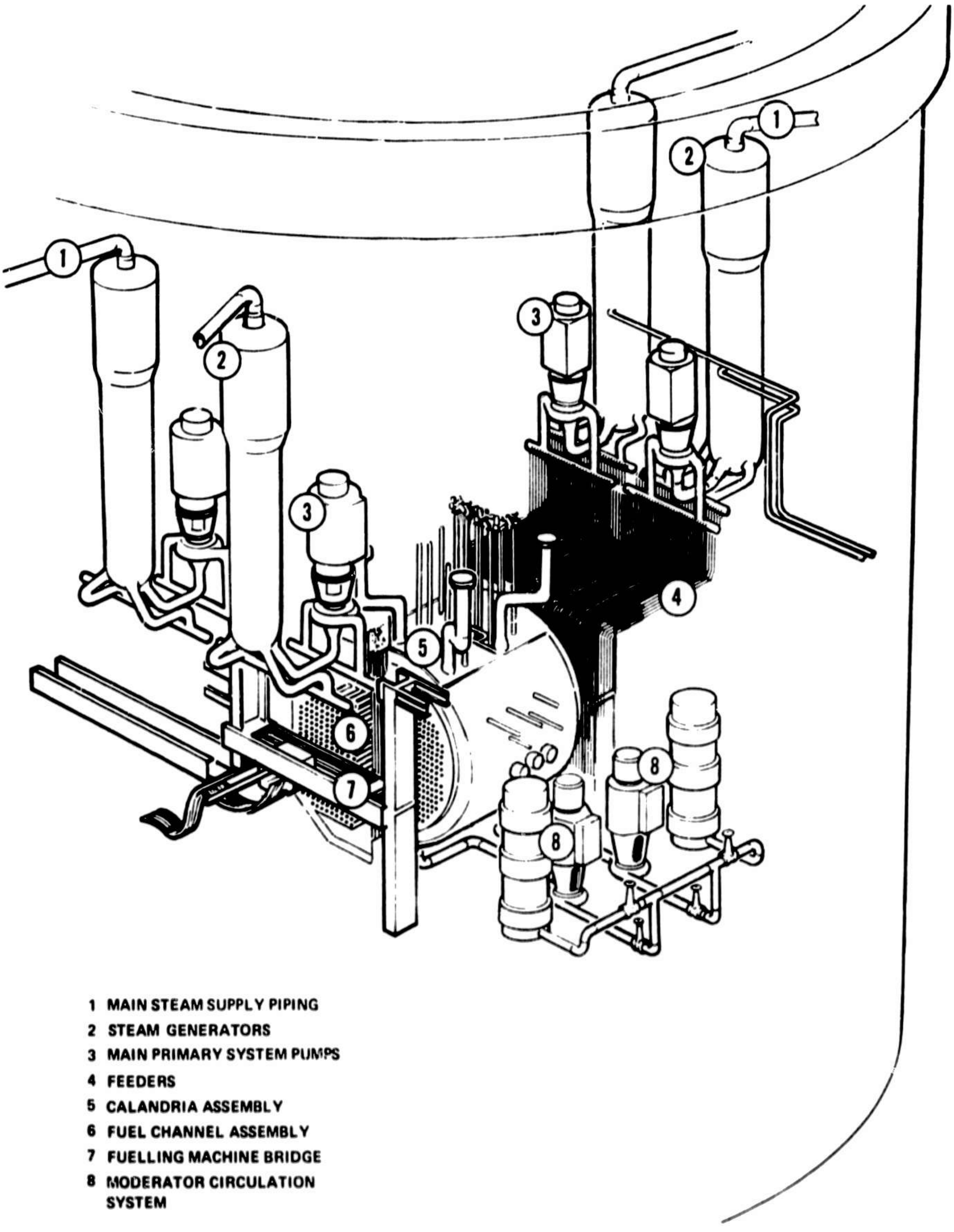


Fig. 11.5 CANDU-PHW Reactor

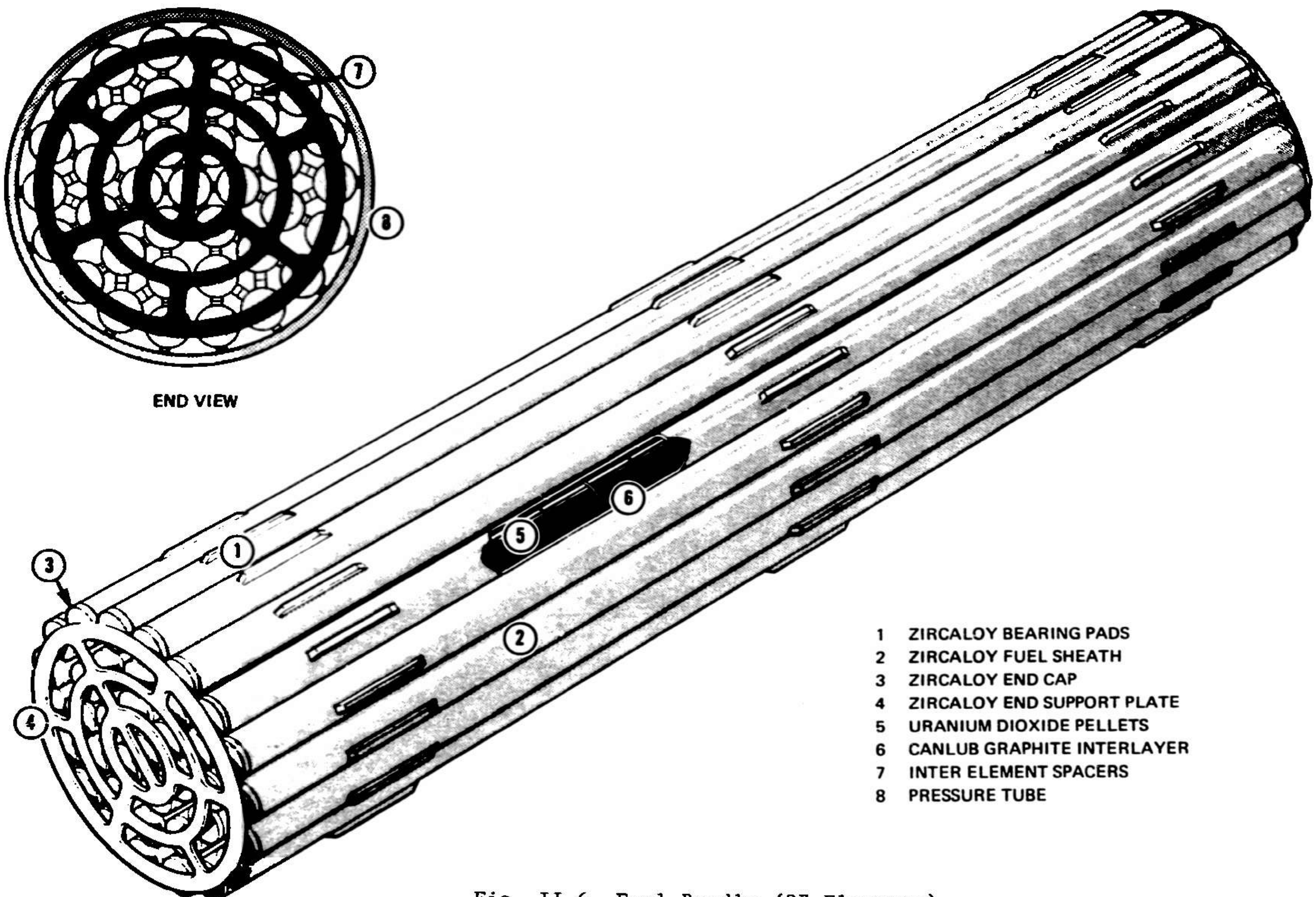
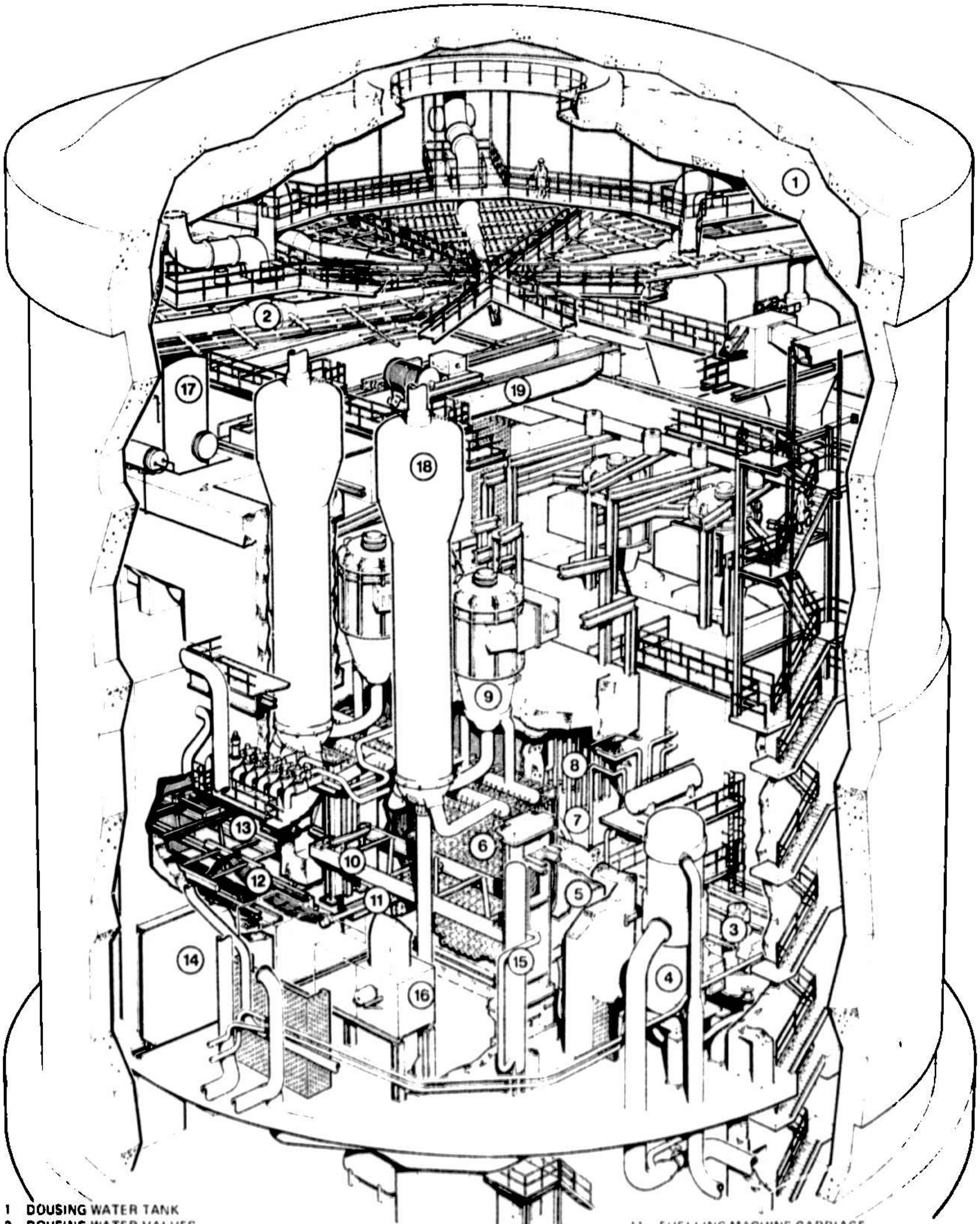
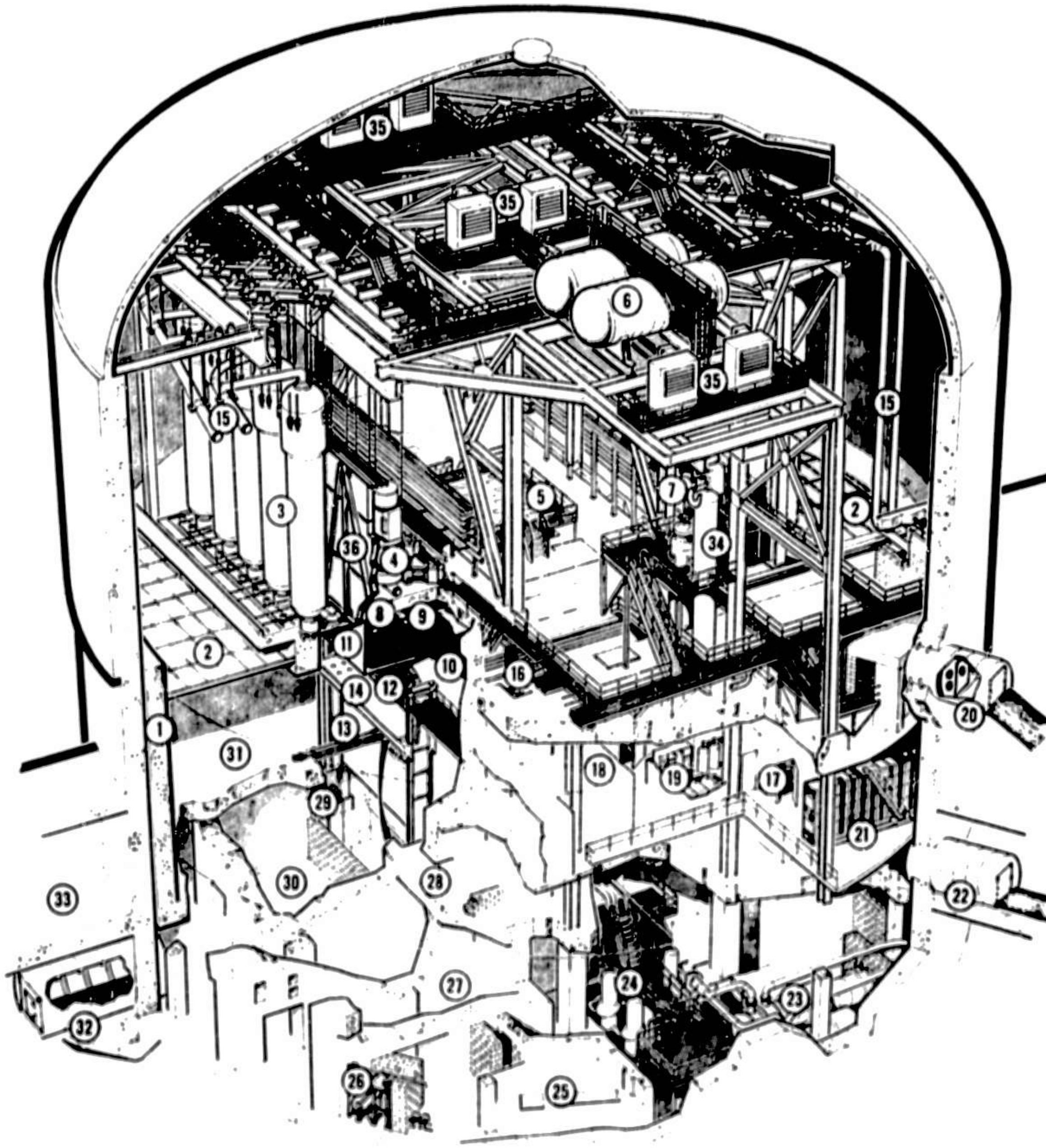


Fig. II.6 Fuel Bundle (37 Elements)



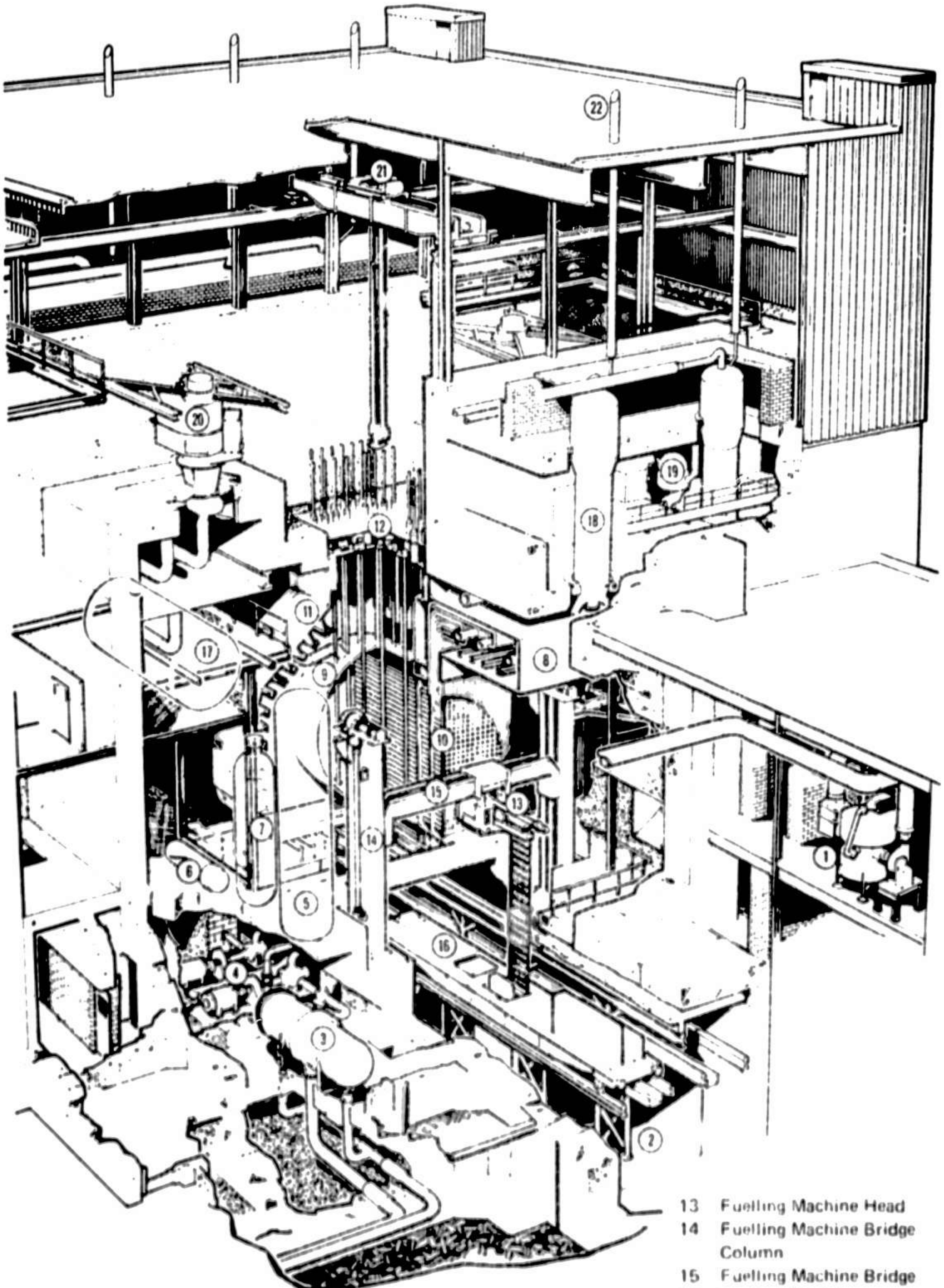
1 DOUSING WATER TANK
2 DOUSING WATER VALVES
3 MODERATOR PUMP
4 MODERATOR HEAT EXCHANGER
5 FEEDER CABINETS
6 REACTOR FACE
7 REACTOR
8 REACTIVITY MECHANISM
9 PRIMARY HEAT TRANSPORT SYSTEM PUMP
10 FUELLING MACHINE BRIDGE
11 FUELLING MACHINE CARRIAGE
12 FUELLING MACHINE CATENARY
13 FUELLING MACHINE MAINTENANCE LOCK
14 FUELLING MACHINE MAINTENANCE LOCK DOOR
15 END SHIELD COOLING WATER DELAY TANK
16 VAULT COOLER
17 PRESSURIZER
18 STEAM GENERATOR
19 STEAM GENERATOR ROOM CRANE

Fig. II.7 Reactor Building Cutaway of a Standard Single-Unit Station



1 PRESSURE WALLS
2 BLOWOUT PANELS
3 STEAM GENERATORS
PRIMARY HEAT TRANSPORT PUMPS
CONTROL AND SHUT-OFF RODS
6 FEED WATER RESERVE TANKS
7 BOILER ROOM CRANE
8 PRIMARY HEAT TRANSPORT REACTOR OUTLET HEADER
9 PRIMARY HEAT TRANSPORT REACTOR INLET HEADER
10 FEEDER PIPES
11 FEEDER INSULATION CABINET
12 REACTOR END FITTINGS
13 FUELLING MACHINE HEAD
14 FUELLING MACHINE BRIDGE
15 MAIN STEAM SUPPLY PIPES
16 PIPE CHASE
17 INSTRUMENTATION ROOM (WEST)
18 D_2O COLLECTION ROOM
19 ZONE CONTROL SYSTEM ROOM
20 BOILER ROOM AIRLOCK
21 REACTOR CONTROL DISTRIBUTION FRAME
22 MAIN EQUIPMENT AIRLOCK
23 MODERATOR HEAT EXCHANGERS
24 MODERATOR PUMPS
25 MODERATOR AND ION EXCHANGE COLUMNS
26 SPENT RESIN DRYING TANK
27 FUELLING MACHINE AUXILIARIES ROOM (EAST)
28 FUELLING MACHINE VAULT DOORWAY
29 FUEL TRANSFER PORT
30 FUELLING MACHINE SERVICE ROOM (EAST)
31 FUELLING MACHINE VAULT (EAST)
32 FUELLING MACHINE AIRLOCK
33 REACTOR AUXILIARIES BAY
34 BLEED CONDENSER AND BLEED COOLER
35 BOILER ROOM COOLING UNITS
36 SHIELDING WALL

Fig. II.8 Reactor Building Cutaway of a Multi-Unit Station with Vacuum Containment (Steam Generators and PHTS Pumps Inside the Containment)



1 Air Dryers
2 Fuelling Machine Duct
3 Moderator Heat Exchanger
4 Moderator Pumps
5 Pressurizer
6 Bleed Cooler
7 Bleed Condenser
8 Feeder Cabinet
9 Calandria
10 End Fittings
11 Shield Tank
12 Reactivity Mechanism
13 Fuelling Machine Head
14 Fuelling Machine Bridge Column
15 Fuelling Machine Bridge
16 Fuelling Machine Transport Trolley
17 Heavy Water Storage Tank
18 Steam Generator
19 Preheater
20 Primary Pumps
21 Bridge Crane
22 Safety Blow-off Valves

Fig. II.9 Reactors Building Cutaway of a Multi-Unit Station with Vacuum Containment (Steam Generators and PHTS Pumps Partially Outside the Containment)

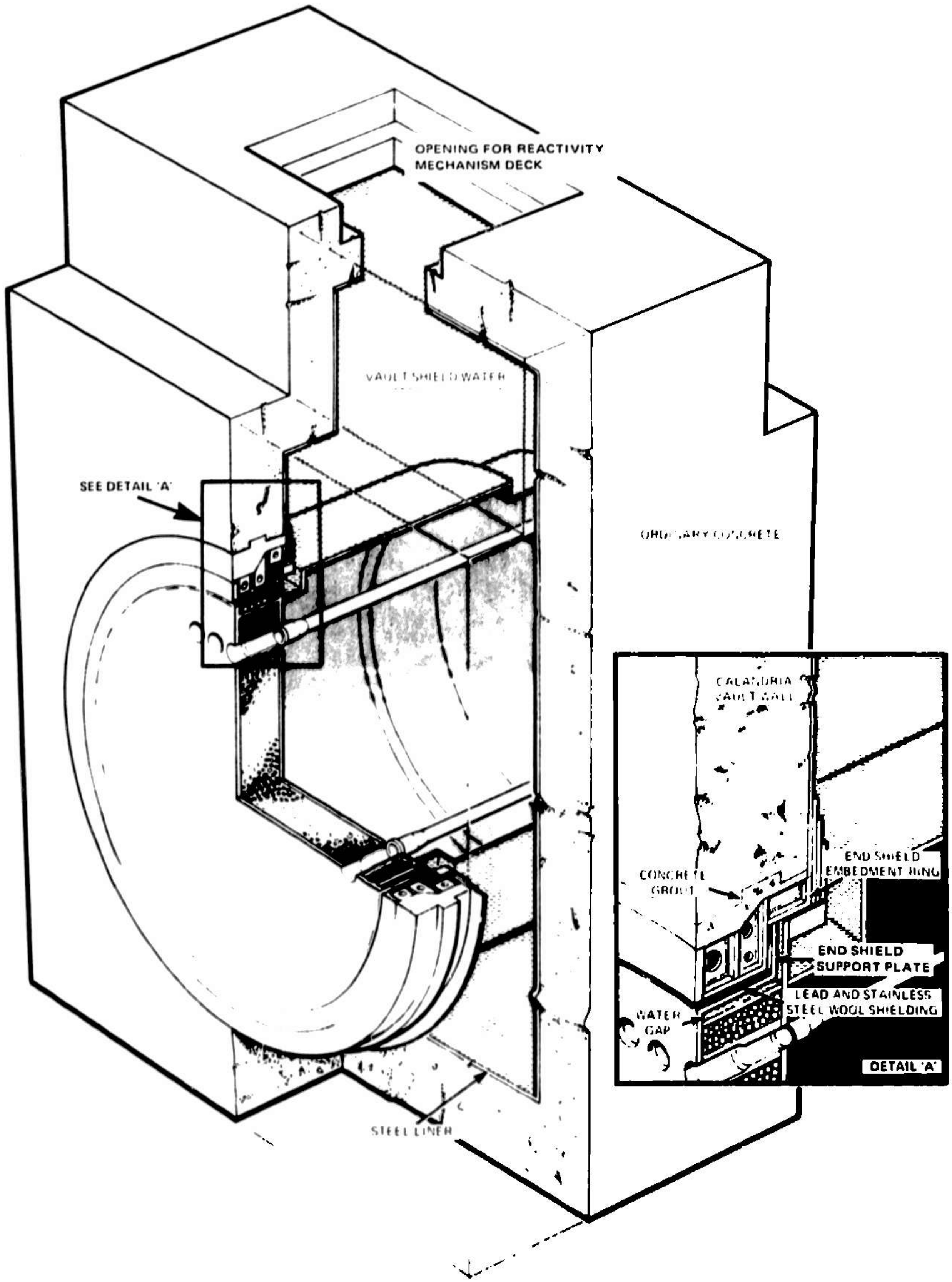


Fig. 11.10 Concrete Calandria Vault

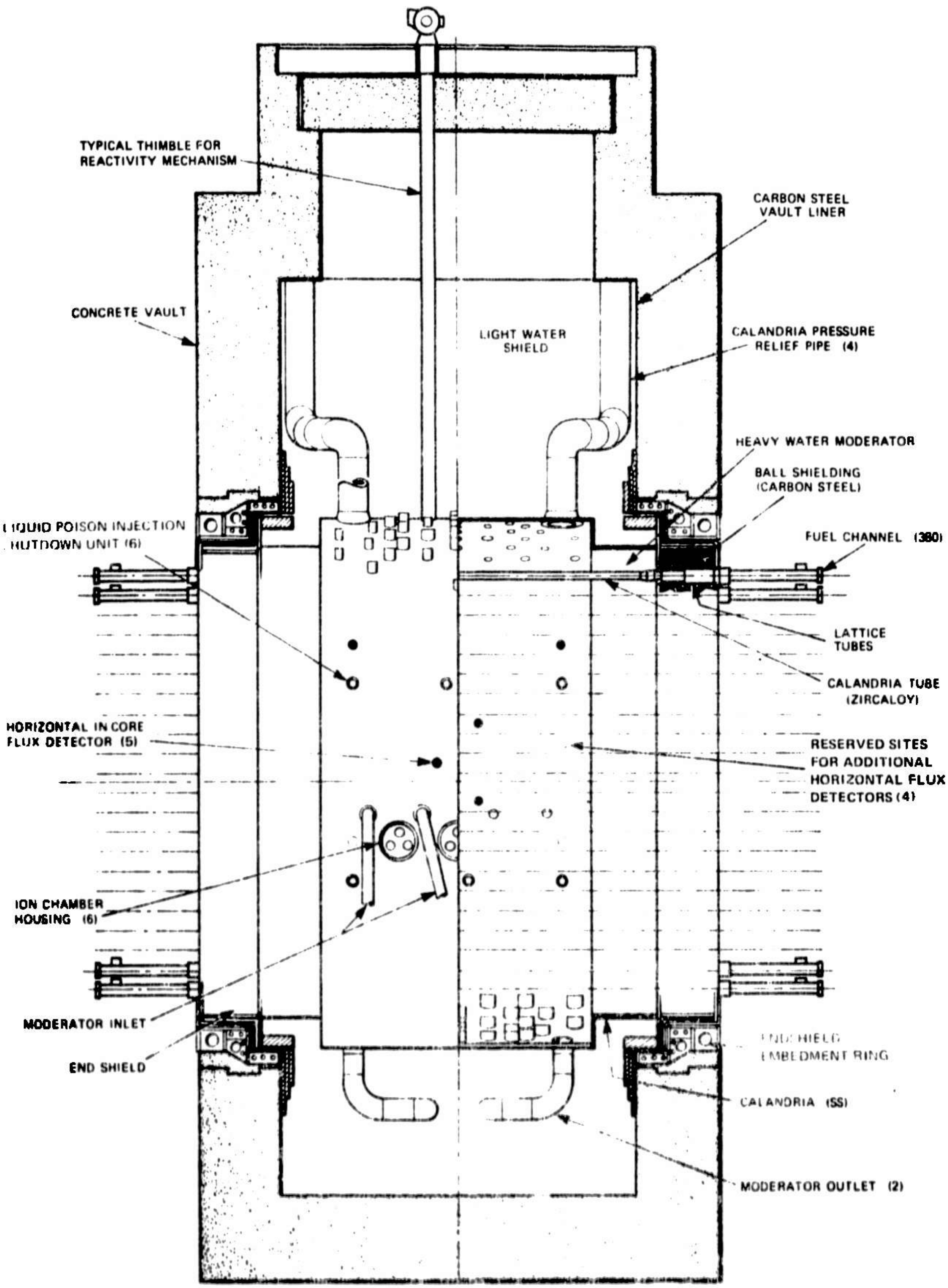
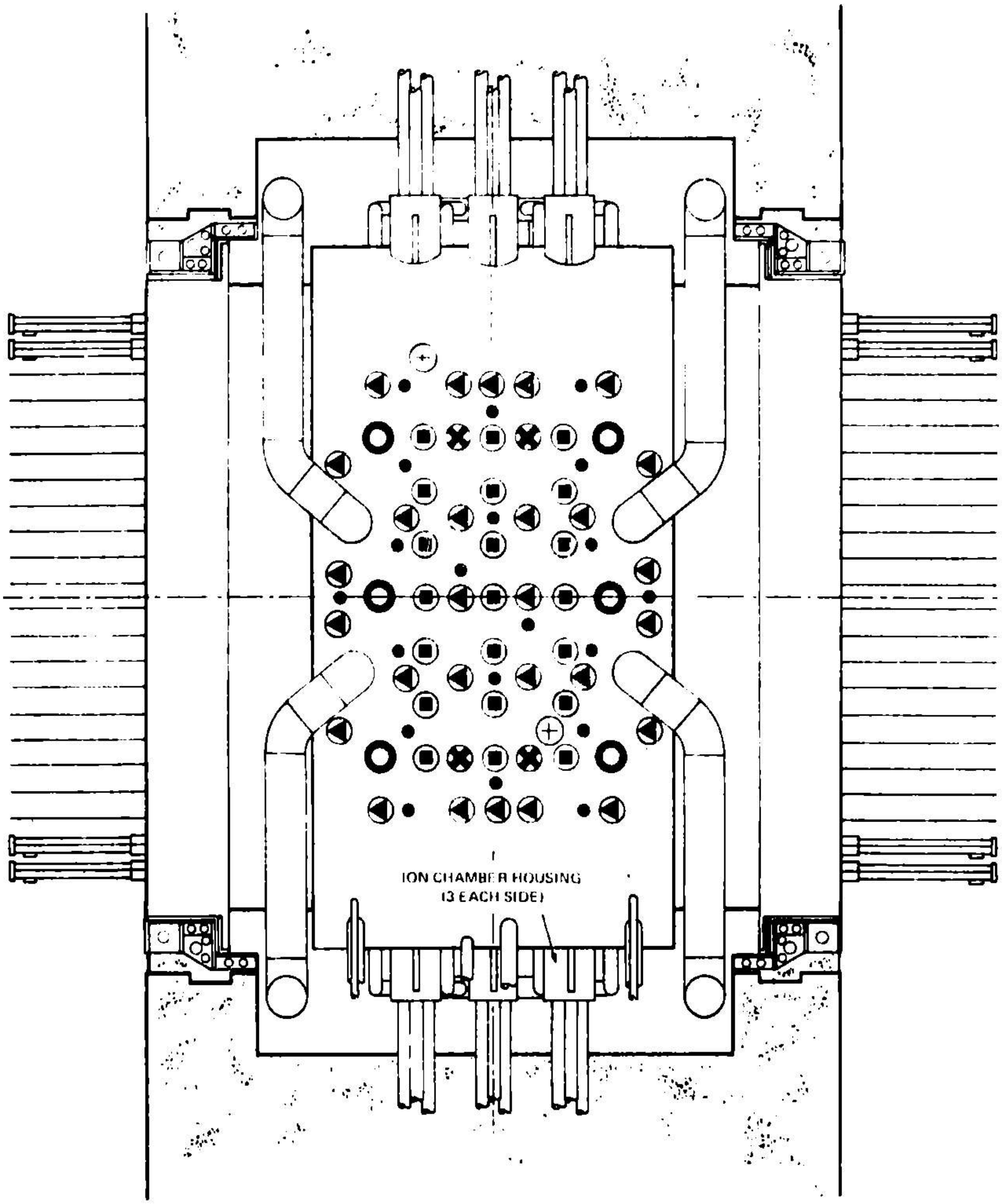


Fig. II.11 Reactor General Assembly (Section)



● VERTICAL FLUX DETECTOR (20)

ADJUSTER (21)

SHUTOFF UNIT (28)

LIQUID ZONE CONTROL (6)

SOLID CONTROL ABSORBER (4)

(+) VIEW PORT (2)

Fig. II.12 Reactor General Arrangement - Plan

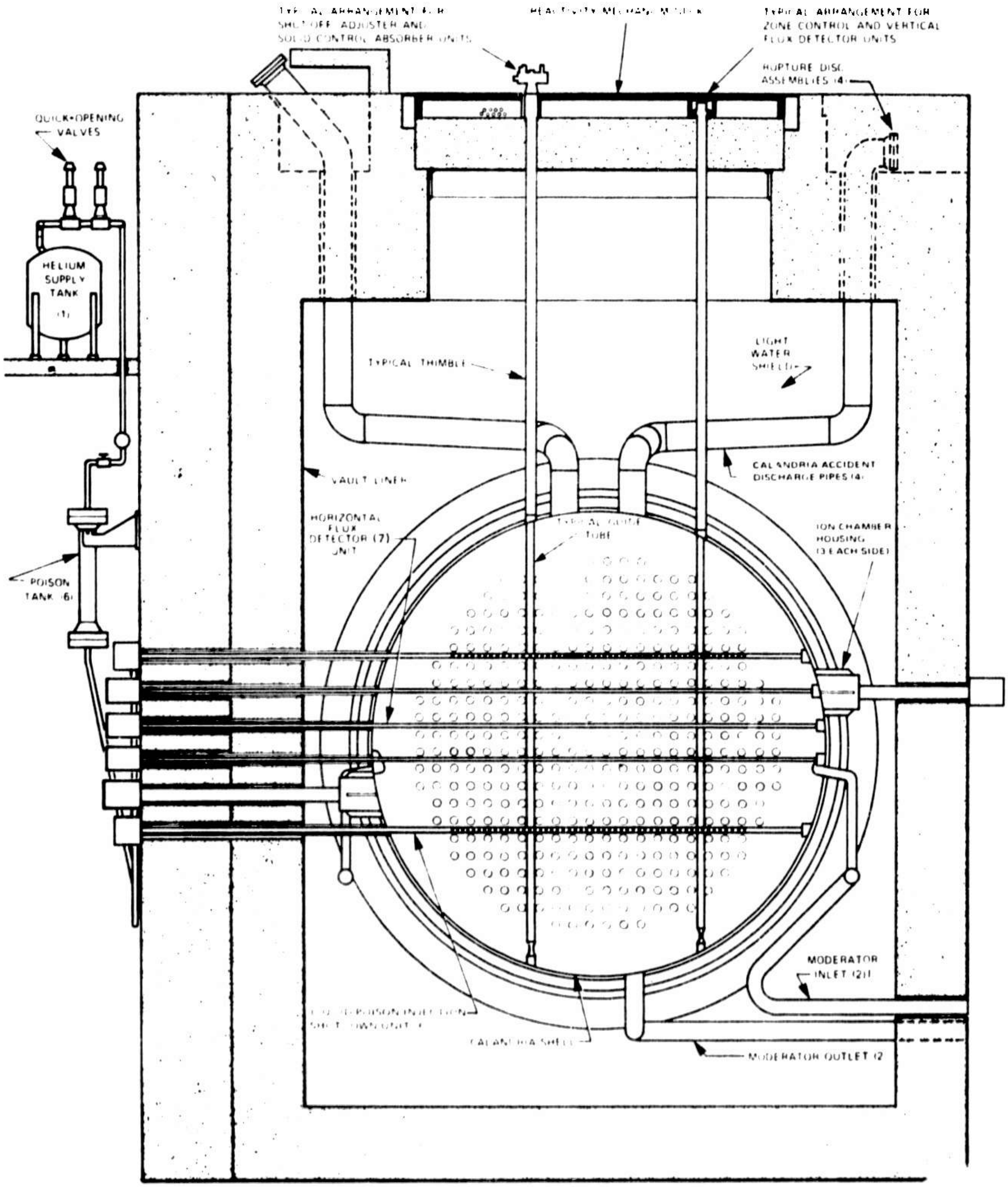


Fig. II.13 Reactor General Arrangement - Elevation

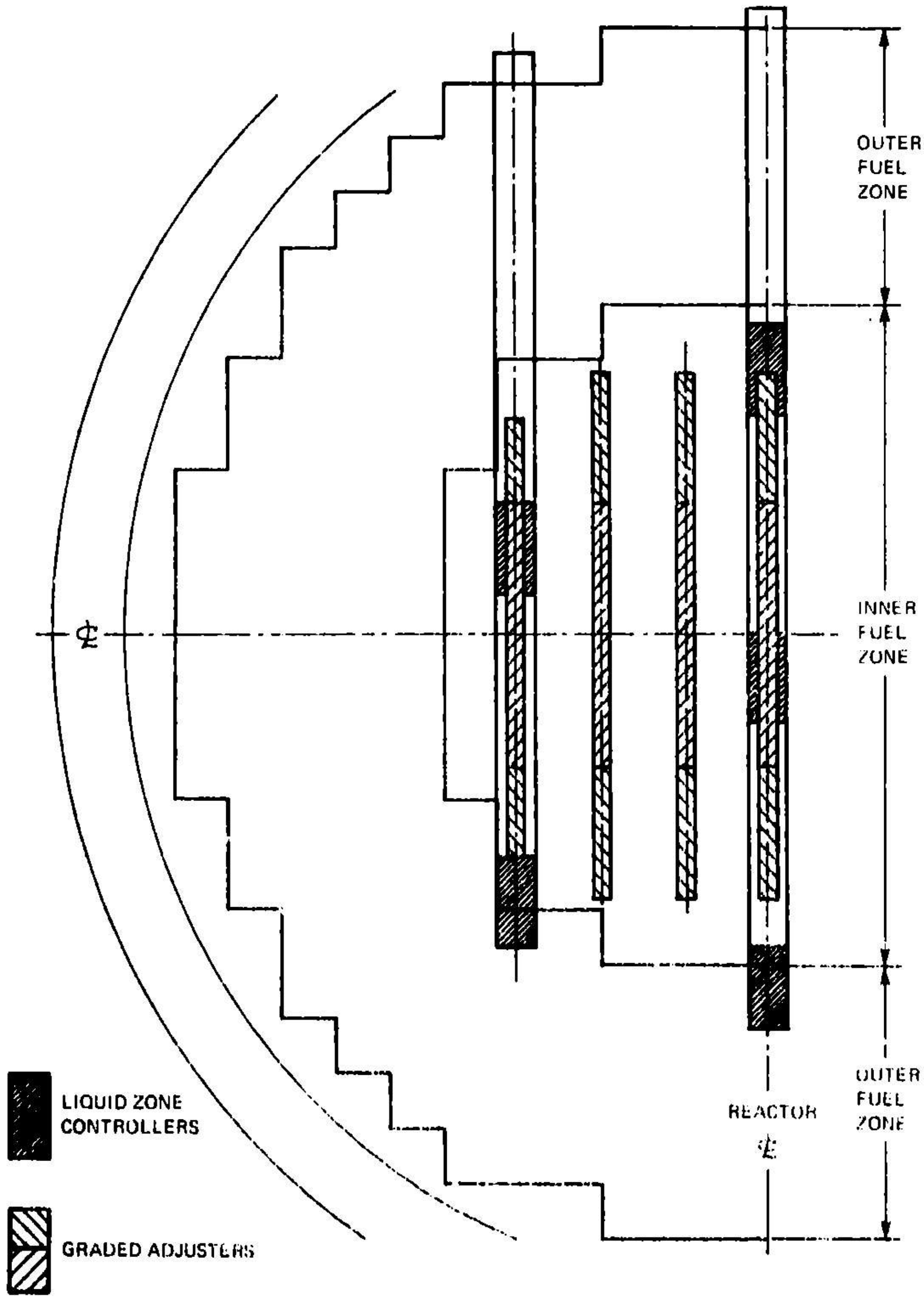


Fig. II.14 Reactivity Device Layout

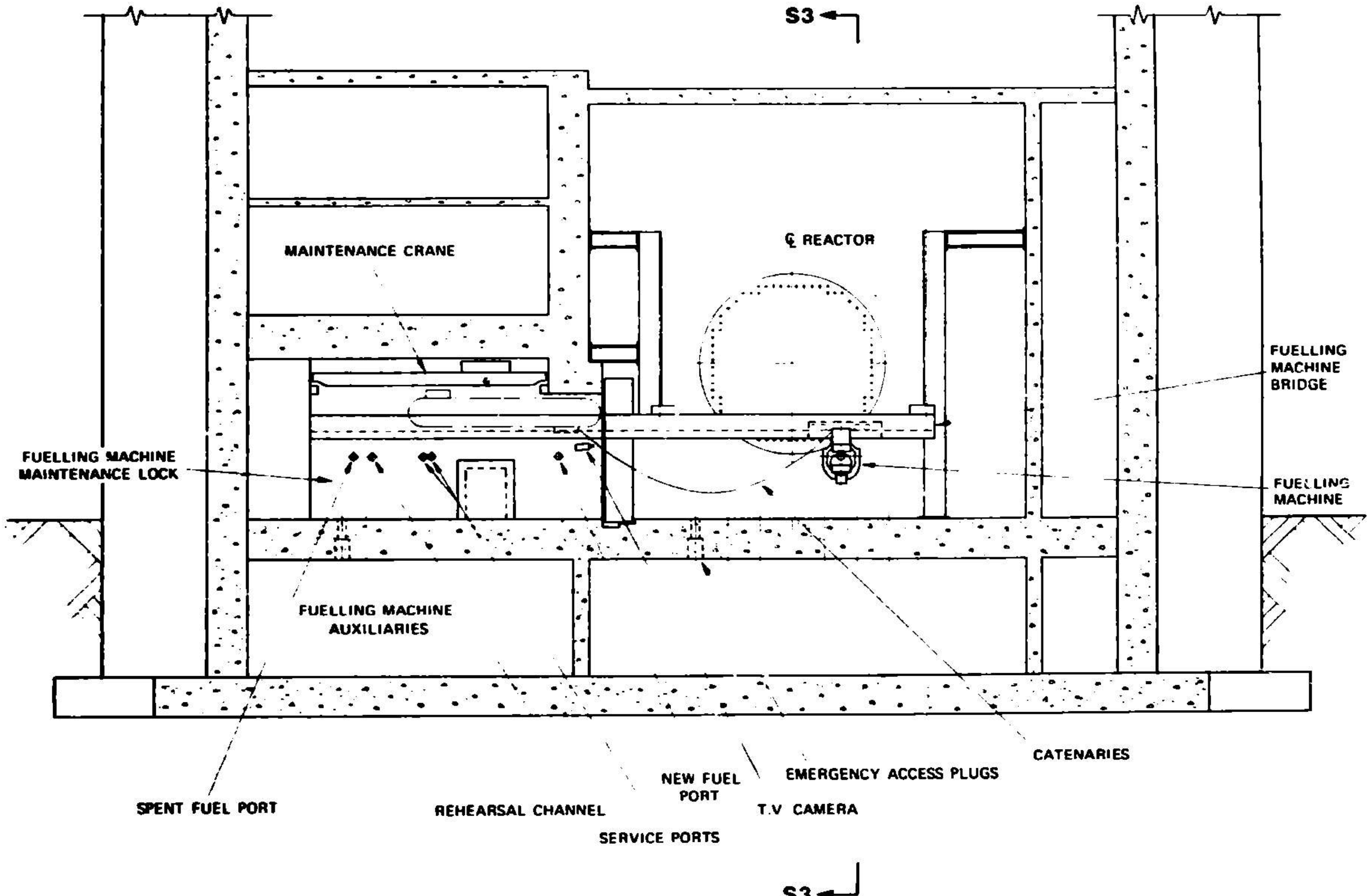


Fig. II.15 Fuel Handling System - Section S2

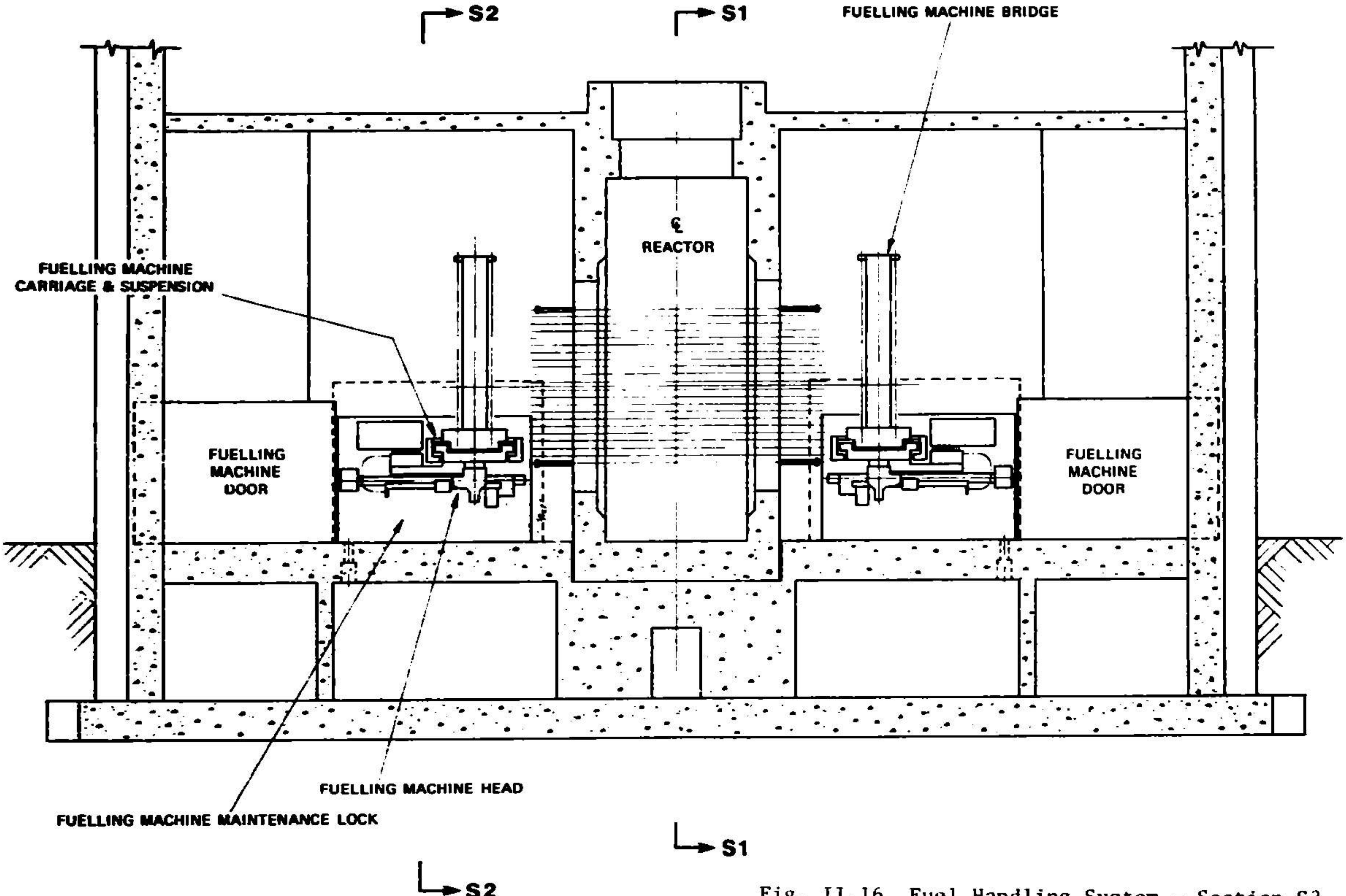


Fig. II.16 Fuel Handling System - Section S3

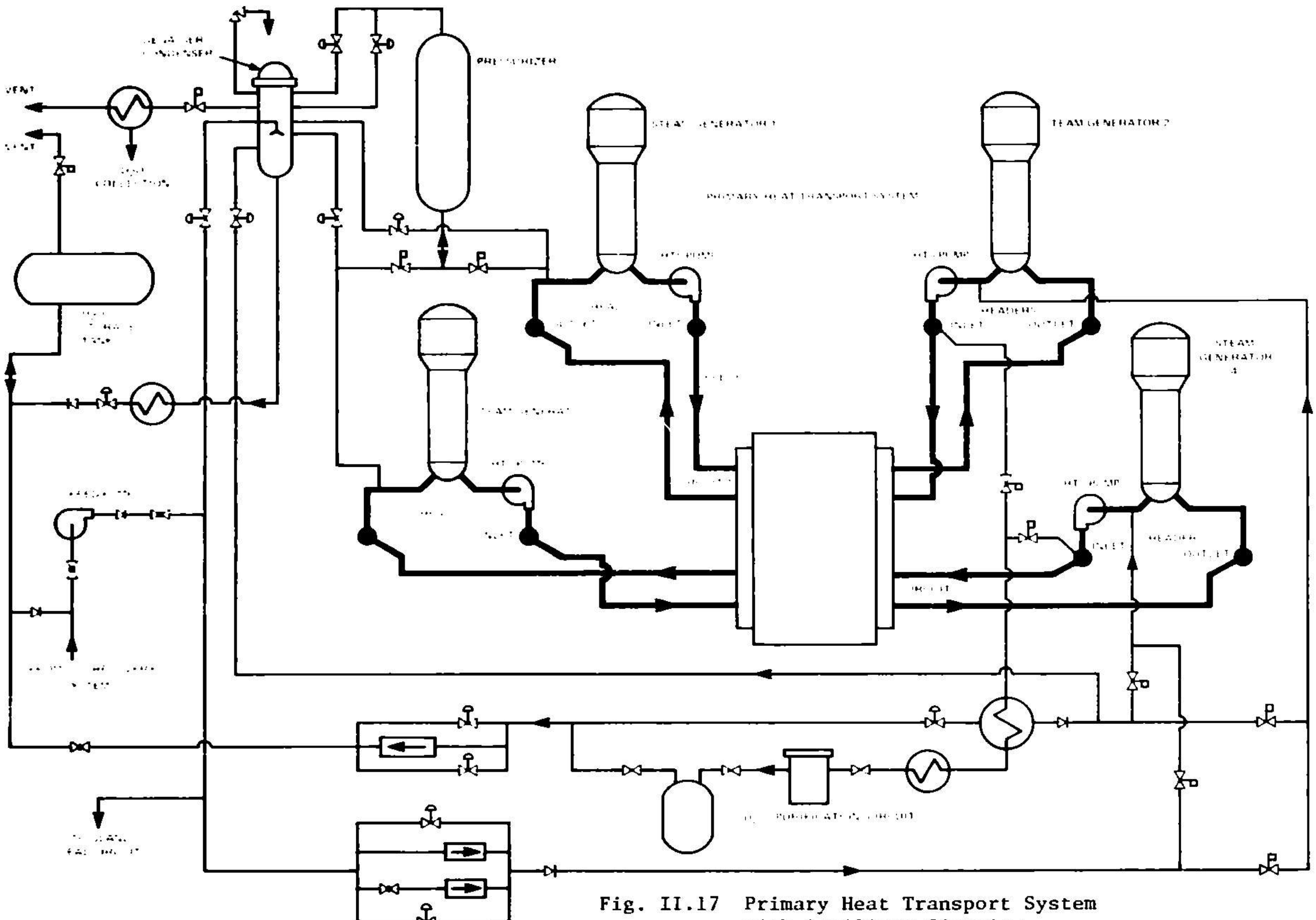


Fig. II.17 Primary Heat Transport System with Auxiliary Circuits

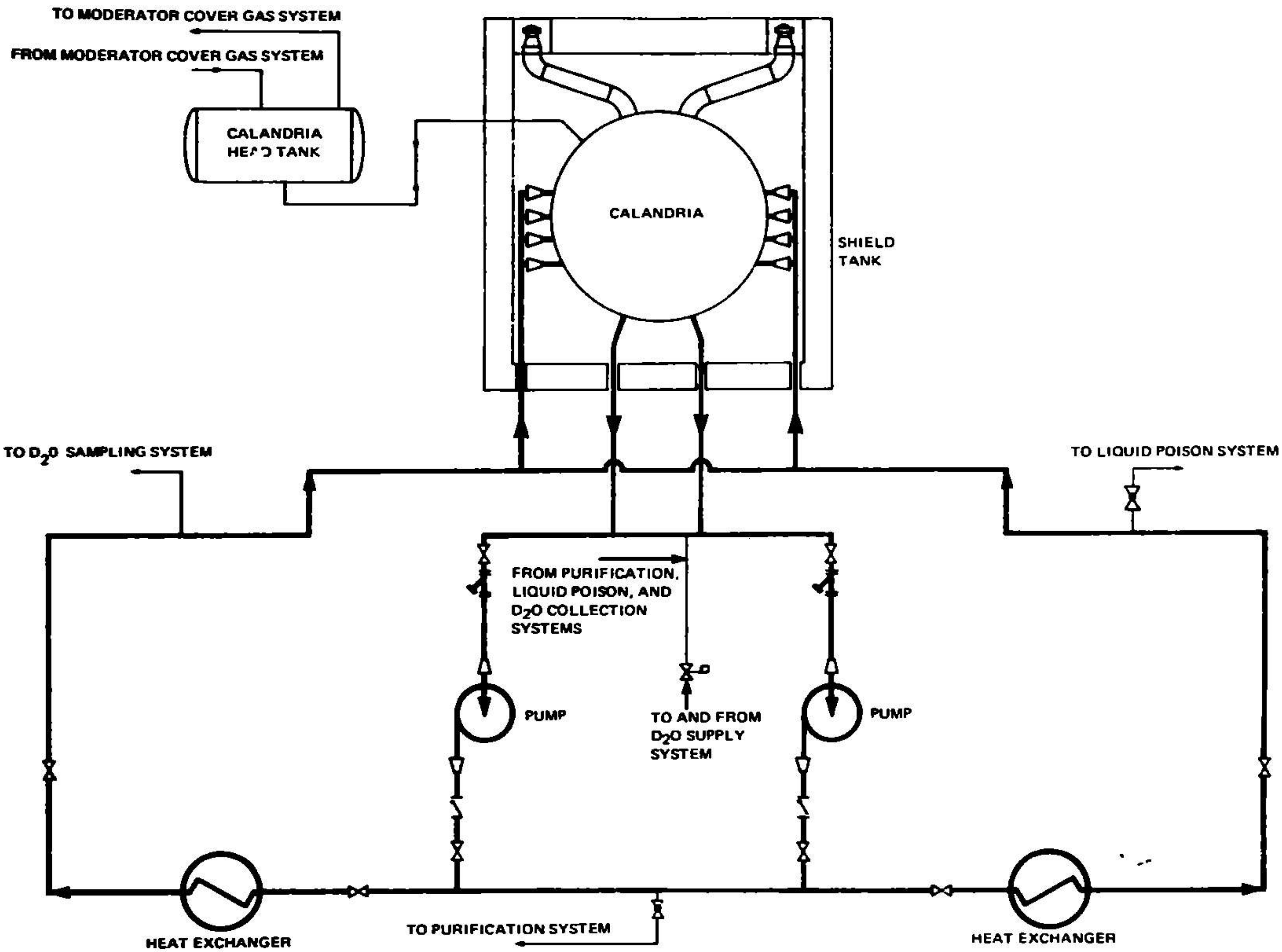


Fig. II.18 Main Moderator System

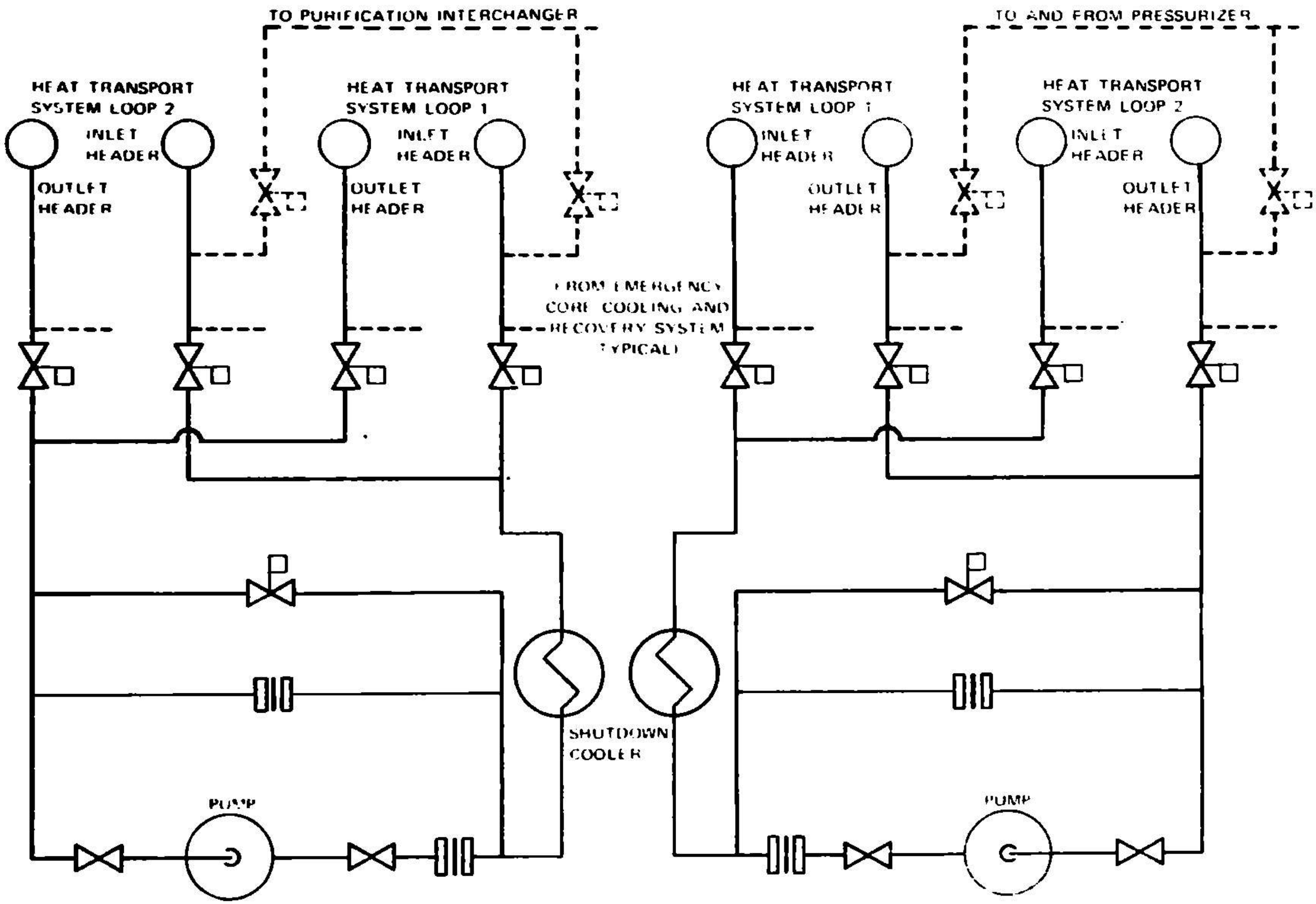


Fig. II.19 Primary Heat Transport System
Shutdown Cooling System

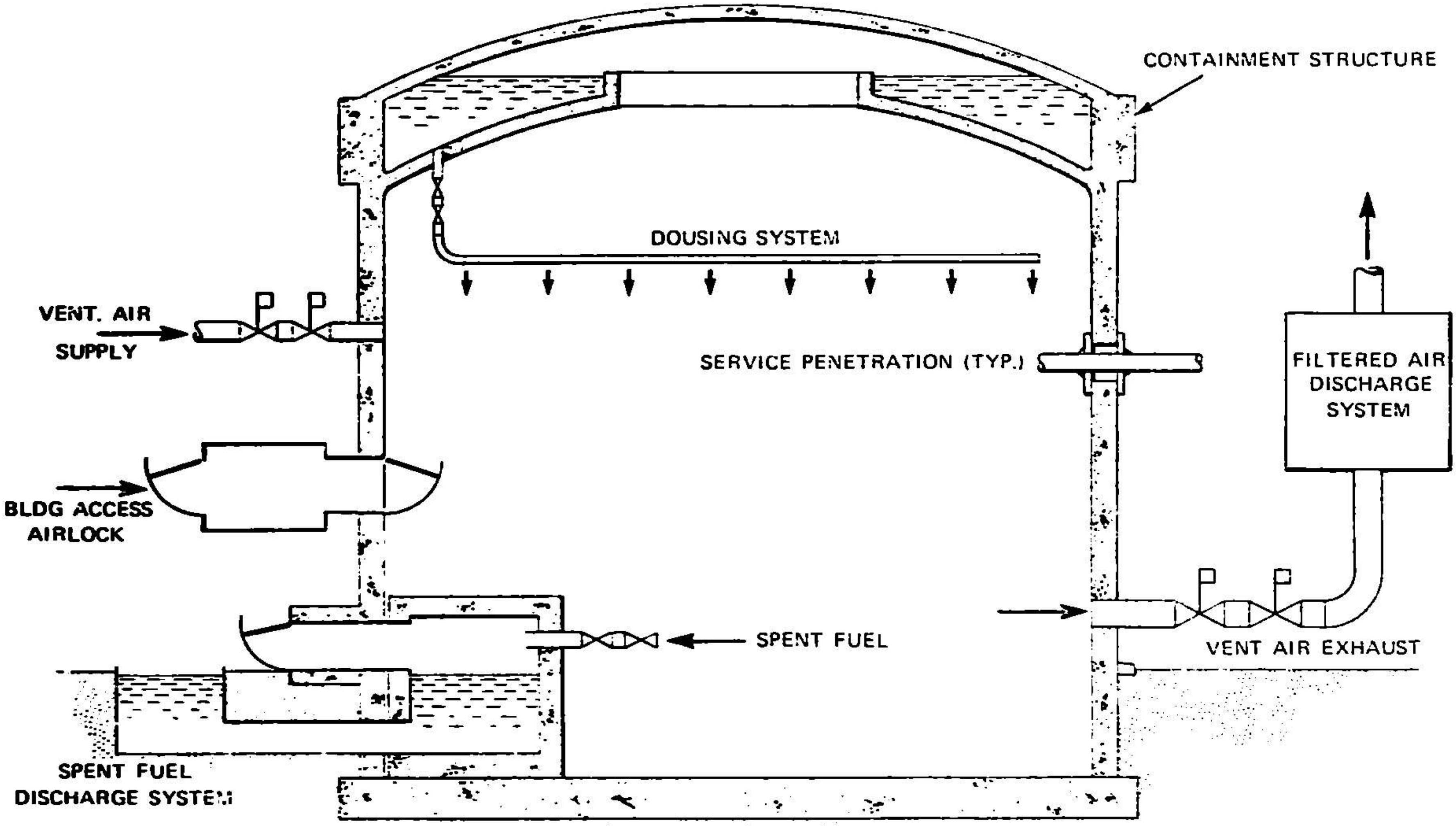


Fig. II.20 Single-Unit Containment System

III. Safety-Related Intrinsic Characteristics of the CANDU NSSS

Table III.1 gives a list of some of the important safety-related characteristics intrinsic to CANDU reactors. Some of the implications of these characteristics will be briefly discussed:

The fact that the primary coolant pressure boundary of CANDU reactors consists mainly of tubes is one of the most obvious points of difference with LWRs. This makes it possible to design the primary heat transport system (PHTS) for large tolerance to rupture of any single primary coolant pressure bearing component. It should be noted, however, that the pressure tubes are exposed to the full neutron flux, so that some degree of embrittlement may be expected with time. This may require some attention in that the probability of pressure tube failure has to be shown to be very low at any point in the life of the reactor. In LWRs, none of the stress-bearing components in the primary coolant boundary are exposed to the full neutron flux. It should be kept in mind, however, that the probability of the occurrence of a sudden large-size break in a pressure tube in a CANDU reactor is extremely low, in view of the following considerations:

(1) The tube-wall thickness is much smaller than the critical crack size for catastrophic failure so that leakage will precede tube rupture ("leak-before-break")[15-19];

(2) A leak of a pressure tube can be detected quickly (by means of the surveillance system analyzing the gas contained in the annular space between pressure tubes and calandria tubes, as well as by means of an ultrasonic sound pick-up system installed on on the head of the fueling machine) thus allowing ample time for corrective action;

(3) The pressure tubes and their end-fittings can be inspected while the reactor is in operation by means of ultrasonic probes used in conjunction with the on-load fueling machine, thus providing an up-to-date overview of the state of the pressure tubes; and

(4) Although the pressure tubes are designed to serve for the entire life time of the plant, they can be replaced with relative ease, thus permitting early elimination of tubes showing any signs of faults.[20]

The subdivision of the CANDU core region into separate power channels has a number of implications, some of which are listed in Table III.1 under items (2) through (4). One of the primary objectives of the current U.S. LWR safety program is to prove timely re-establishment of cooling for all core regions by the emergency core cooling system (ECCS) following a loss of coolant accident (LOCA). In the LWR, penetration of emergency coolant into the core is hampered by generation of steam (which tends to expel the coolant from the core region as it enters), as well as by bypassing of the core by the ECCS coolant. Similar problems may exist to some extent for the emergency coolant injection system (ECIS) of CANDU reactors; however, the following considerations support the position that the emergency cooling issue is less critical for CANDU reactors:

(1) The simple configuration of the individual fuel channels tends to facilitate coolant delivery to all core locations (no downcomer region, etc.);

(2) Experiments aimed at verifying the performance of the emergency coolant injection system (ECIS) for CANDU reactors appear to be simpler and more conclusive than those aimed at verifying ECCS performance for LWRs in that the results of the former are less scale-dependent.

(3) The correct performance of the ECIS does not constitute for CANDU reactors the final defense against core meltdown in case of LOCA, as is the ECCS for LWRs. Canadian analyses, supported by experiments, indicate that a LOCA combined with ECIS failure, though resulting in substantial fuel damage (including partial melting of the cladding) and some deformation of the pressure and calandria tubes, does not result in fuel melting. The decay heat can be removed by conduction through the walls of the pressure and calandria tubes into the moderator, and rejected by the moderator cooling system.

For small breaks in the primary cooling system, long-term cooling in CANDU reactors is provided by natural circulation with the steam generators acting as heatsink; for a large-break LOCA, long-term cooling is provided by once-through convection to the break with the emergency coolant recovery system serving as heatsink.

CANDU reactors have a positive void-reactivity coefficient. However, the total excess reactivity available in a natural-uranium-fueled reactor system with on-load refueling is rather limited; it is furthermore possible to limit the rate of reactivity insertion by a LOCA by subdividing the primary cooling system into separate subsystems. This approach was followed for the Pickering nuclear power station, which has a primary cooling system consisting of two completely separate subsystems. For the Bruce nuclear power station, this option was not followed, although the standard 600 MWe CANDU plant has again a PHTS consisting of two independent subsystems. The total reactivity introduced by completely voiding all pressure tubes in the core region at nominal operating conditions is $\sim$ 1.5\$.

It is of interest to mention here that the CANDU reactors are not alone in having a pressure-dependent reactivity effect, capable of positive reactivity insertion under accident conditions. LWRs also have pressure dependent positive reactivity effects: (1) Pressure increase in BWRs results in void collapse and reactivity insertion, and (2) control rod ejection in PWRs and BWRs results in reactivity insertion.

An important intrinsic safety-related characteristic of the CANDU reactor is the fact that all neutron control devices are installed in the low-pressure moderator region where, in case of a postulated LOCA due to a break in the headers or the feeders, they are not subjected to potentially severe hydraulic forces (as is true for LWRs). Furthermore, the relatively open core lattice of the CANDU reactor permits complete separation between control and protection functions, also for the neutron poison devices. There are, apart from the control system, two completely independent reactor safety shutdown systems in a CANDU reactor, one of which makes use of neutron poison rods and one of which is based on gadolinium injection into the moderator.

The on-load refueling capability, listed in Table III.1 under point (12) has a number of safety-related implications; one is that the fueling machine becomes part of the PHTS during the refueling operation, necessitating appropriate seismic design for the site in question which has been provided; a second is that on-load refueling results in a reactor with relatively small reactivity requirements for the control system.

TABLE III.1 Some Important Safety-Related Intrinsic Characteristics of Heavy-Water-Moderated Pressure-Tube Reactors

	Characteristics	Safety Implications
1.	The pressure tubes (which are part of the primary coolant pressure boundary), traverse the active core region.	-Stress-bearing components of the coolant pressure boundary are subjected to the full neutron flux.
		-Rupture of a pressure tube in the core region has to be analyzed as part of the safety evaluation.
		-The probability for tube-to-tube failure propagation in the core region must be very low.
2.	The pressure tubes (having a relatively small wall thickness) have leak-before-break characteristic.	-The probability of a sudden large-size break in a pressure tube is very small, because the tube will first develop a leak.
3.	The pressure tubes are surrounded by calandria tubes, creating a gas-filled annular space between the tubes.	-A crack in a pressure tube, resulting in primary coolant leakage, is easily detected by means of the surveillance system analyzing the gas contained between pressure tubes and calandria tubes (leaks can also be detected by means of the ultrasonic detection system; see point 12).
4.	The core is subdivided in separate fuel channels having individual coolant supply.	-The primary cooling system can be subdivided into a number of subsystems, thus limiting complete blowdown to only a part of the core in case of a loss-of-coolant accident (LOCA).
		-LOCA is mitigated by hydraulic resistance in piping, in part due to the figure-of-eight layout of the PHTS.
		-The ECIS is capable of delivering emergency coolant to all core locations with low probability of performance failure.
		-The simple configuration of the power channels (pressure tube + fuel) allows relatively easy testing of ECIS performance (scaling is relatively easy).
		-Failed fuel can be easily detected and located.

TABLE III.I (Contd.)

	Characteristics	Safety Implications
5.	Large inventory of cold moderator, having redundant cooling system with capability of removing decay heat.	-Large dispersed heat sink in core region.
		-Moderator heat capacity and cooling system serve as diverse back-up system for ECIS.
6.	Moderator region is surrounded by large light water shield region, having large heat capacity and redundant cooling system with capability $\sim$ 0.3% of nominal power.	-Provides additional heat sink close to core region, which could serve as back-up system for ECIS and moderator cooling system.
7.	Total excess reactivity is small for natural-uranium fueled equilibrium core.	-Relatively mild power excursions due to accidental reactivity insertions.
8.	Power-reactivity coefficient at nominal power level is close to zero, and may be slightly positive.	-Power transients due to uncompensated reactivity insertions would tend to be not self-limiting.
9.	Void-reactivity coefficient is positive.	-LOCA leads to a reactivity increase.
		-Under-cooling transients lead to reactivity increase due to boiling in power channels.
10.	Mean neutron lifetime is $\sim 10^{-3}$ sec, i.e., $\sim$30 times larger than for LWRs.	-Power transients tend to be, for the same reactivity insertion, less severe for CANDU reactors than for LWRs.
11.	The neutron poison devices for control and safety shutdown are installed in the low pressure moderator region.	-There is no pressure-assisted reactivity accident associated with the control or shutdown rods (compare with rod-ejection accident in LWRs).
		-Control rods and safety-shutdown rods are not subjected to hydraulic forces in case of an ex-core LOCA (contrary to what is the case for LWRs).

TABLE III.1 (Contd.)

	Characteristics	Safety Implications
12.	On-load refueling.	-Failed fuel can be easily replaced without necessitating reactor shutdown, thus providing a means for maintaining a low level of radioactivity in the PHTS. -Refueling malfunctions could result in small scale LOCA. -Jamming of fuel subassembly during refueling operation could result in under-cooling incident, affecting a single channel. -Fueling machine becomes part of the PHTS during refueling operation, requiring appropriate seismic design for the site in question which has been provided. -Ultrasonic detection system, installed on the head of the fueling machine, provides a means for in-service inspection of the pressure tubes: (a) ultrasonic probes permit volumetric inspection of the pressure tubes for early detection of crack formation, (b) ultrasonic sound pick-up permits early detection of leaks. -On-load refueling results in a reactor system with relatively low control reactivity requirements.
13.	Burnup of fuel is low (< 10,000 MWD/ton).	-Fission product inventory is relatively small.
14.	Inventory of tritium relatively large.	-Requires special attention (however, the major part of the tritium inventory is in the low-pressure moderator region; furthermore economic considerations do not allow loss of significant amounts of heavy water).

IV. Safety-Related Non-Intrinsic Characteristics of the CANDU NSSS and the Balance of Plant

Table IV.1 gives a list of some safety-related non-intrinsic characteristics of the NSSS and the balance of plant. It should be kept in mind that design changes have been, and are being, introduced from plant to plant so that some of the characteristics listed may apply to only a few units and not necessarily to others. As an example, there exists a greater degree of separation between the containment buildings of the individual units for the Pickering Station than for the Bruce Station. The Bruce Station has four permanently interconnected containment buildings, which can be connected to the vacuum building only as a group. Single-unit containment systems do not use a vacuum building.

There exists complete separation between control and protection functions, even to the point of having separate neutron poison devices. Many control functions are performed by computers, including plant start-up, refueling, etc. All safety-related functions are performed by hard-wired circuitry, which is fully monitored and alarmed within each safety system. In addition, some safety-related data are also handled by the computers for reasons of operational convenience (data processing and recording only; no feedback).

The use of computers for certain control functions may not conform in all details to current US industrial practice but appears to be not in conflict with current U.S. licensing regulations and criteria. In CANDU plants, redundancy has been provided for computer-operated control functions. Two nearly identical computers, linked only by a data channel, are operated in a main and standby configuration.

It appears justified to expect that accommodation to current U.S. licensing regulations and criteria, e.g., regarding redundancy (single failure criteria, etc.) of the various auxiliary systems and engineered safeguards of the CANDU reactor can be accomplished without great difficulty by redesign of some of the peripheral systems of the NSSS.

TABLE IV.1 Some Safety-Related Non-Intrinsic Characteristics of Current CANDU-PHW Reactors

	Characteristics	Safety Implications
1.	Use of computers for:	
	-Power control (both overall power and zonal power).	-Automatic zonal power control reduces the probability for local fuel damage in the core.
		-Regional core protection system protects against low-probability malfunction of zonal power control system.
	-Control of on-line refueling operation.	-Malfunction of fueling machine could possibly result in PHTS leak.
	-Annunciation and event recording.	-Results in improved plant surveillance.
	-Recording of selected process variables.	-Results in improved plant surveillance.
2.	Protective functions are kept strictly separated from control functions and are performed by means of hard-wired circuitry.	-This limits considerably the safety implications of malfunction of the control system.
3.	Use of in-core flux sensors in the reactor shutdown systems (SDS1 and SDS2).	-Provides protection against local core damage, due to, e.g. xenon power oscillation and/or zonal power control malfunction.
4.	Use of separate vacuum building for containment system.	-Provides subatmospheric conditions in the containment following a LOCA.
		-Makes it possible to keep containment spray function away from NSSS.
		-May increase sensitivity to seismic events (e.g., by rupture of connecting duct, etc.).

TABLE IV.1 (Contd.)

Characteristics	Safety Implications
5. Use of active components for the vacuum building (valves, vacuum pumps, etc.).	-Valves between the vacuum building and the duct are pressure-actuated by slight overpressure in the duct, and do not require an additional outside energy source. -Valves could, if necessary, be replaced by rupture diaphragms. -Reliance on vacuum pumps could be reduced (e.g., by use of steel liner).
6. Level of redundancy of various auxiliary systems and engineered safeguards systems, such as: - auxiliary feedwater supply - ECIS - containment cooling system - moderator cooling system - emergency power supplies, etc.	-Could be modified for construction in the U.S., if found in variance with current U.S. licensing regulations and requirements.

V. Principal Aspects of the Canadian Safety and Licensing Approach

A. General Approach

The Canadian safety approach, from its early inception, has displayed a tendency towards probabilistic risk assessments. The basic idea is that accidents with low probability of occurrence should be allowed to carry larger consequences than accidents with higher probability. In order to formalize this approach, all systems pertaining to a CANDU reactor are subdivided into two classes, namely (1) the process systems, and (2) the safety systems. The first class (process systems) comprises all systems necessary for the normal operation of the plant (PHTS, control systems, electrical systems, etc.), whereas the second class is made up of all safety systems, i.e., the reactor shutdown systems, the ECIS and the containment system. Table V.1 gives the division between process systems and safety systems. A design requirement is that there be separation among safety systems, and between safety systems and process systems.

Accidents are categorized on the basis of whether they are of the single-failure type, i.e., caused by failure of any one of the process systems, or whether they are of the dual-failure type, i.e., caused by failure of any one of the process systems combined with simultaneous and independent failure of any one of the safety systems. It should be emphasized here that, except for the containment system, failure of a safety system in this context is intended to denote unavailability of the entire system (failure of a component in a redundant safety system could still leave the particular safety function intact); for the containment system different degrees of impairment are postulated. The Atomic Energy Control Board (AECB) has established allowable irradiation doses for individuals and for the total population for the two accident categories. The plant designer has to show, for postulated single-and dual-failure accidents, and for the particular plant in the particular site, that the calculated irradiation doses do not exceed the allowable values. Table V.2 gives the limit doses for single individuals and for the population at large for both postulated single- and dual-failure accidents. Table V.2 furthermore gives Canadian criteria for the maximum frequencies allowable for accidents in the single-failure and dual-failure categories. The designer is required to demonstrate that the frequency of occurrence of serious faults in the entire process system is less than 1 in 3 years, and that the unavailability (unreliability) of each safety system is less than 1 in 10^3 years. Safety systems are required to be testable during plant operation. A serious fault in the process system is defined as one which, in the absence of safety systems, would result in a substantial release of radioactive materials to the environment.

The allowable reference doses in Canada for postulated accidents in the single-failure category (i.e., a serious process system failure, with all safety systems performing as intended) are: 0.5 rem whole-body dose and 3 rem to the thyroid due to I-131 for individuals, and 10^4 man-rem whole-body dose and 10^4 man-rem to the thyroid due to I-131 for the entire population. For accidents in the dual-failure category the maximum allowable frequency of occurrence is 1 in 3 x 10^3 years, while the allowable reference doses are: 25 rem whole-body dose and 250 rem to the thyroid for individuals, and 10^6 man-rem whole-body dose and 10^6 man-rem to the thyroid for the entire population (see Table V.2). For the purpose of comparison dose-limits in force in the US are summarized in Table V.3. It is noted that the allowable Canadian reference doses for accidents in the single-failure category (which include loss-of-coolant accidents of the maximum size) are smaller by a factor of 50 and 100, respectively, for whole-body exposure and thyroid exposure, than those allowable in the US for a similar accident under 10 CFR 100. On the other hand, however, it should be mentioned that a larger degree of conservativeness is incorporated in the US in the radiological source term for dose calculations than in Canada (see Chapter V, Section C for further details).

Consideration of postulated accidents in the dual-failure category (as defined in Canada) is not a requirement in the US-NRC licensing procedure. However, such dual-failure accidents are evaluated in the US on a probabilistic basis (see, e.g., WASH-1400).

Some of the principal Canadian safety criteria are summarized in Table V.4.

B. Hardware-Related Design Requirements

In order to provide protection against events that could induce common-mode failures (fires, airplane crashes, natural phenomena, etc.), the systems have been subdivided into 2 groups, powered from physically separate power sources and provided with separate cooling water supplies. Each group of systems is to have the following capabilities: (1) shut the reactor down to cold conditions, (2) remove decay heat, and (3) provide the operating staff with state-of-reactor information. Table V.5 gives the division of the various systems in the two groups. Figure V.1 gives a schematic overview of the various cooling systems with their power supplies, indicating also the level of their seismic qualification, as required for safety only (economic considerations in some cases impose a higher level of seismic qualification). Table V.6 gives a summary of the actual design classification of the main systems of a current CANDU-PHW plant.

It is Canadian practice to consider in the design of CANDU reactors two levels of severity for seismic events, namely the Design Basis Earthquake (DBE) and the Site Design Earthquake (SDE):

The Design Basis Earthquake (DBE) is defined as, "an artificial representation of the combined effects on the nuclear power plant, at a particular site, of a set of possible earthquakes having a very small probability of exceedence during the life of the plant, and is expressed in the form of response spectra." The DBE is applied to the nuclear power plant structures which are to be seismically qualified to that level of design earthquake. The maximum DBE ground-motion acceleration applied to any CANDU plant under construction today is 0.2 g. The DBE is based on a detailed examination of regional and world tectonics, in addition to an evaluation of historical records, and is expected to have a frequency of $\leq 10^{-3}$ per year, with an overall probability of exceedence of design levels in structures, systems, and equipment qualified to resist the DBE of $\leq 10^{-7}$ per year. In addition, factors of safety of 3 or more are available to ensure that there is no failure of structures, systems or equipment which are essential to nuclear safety following a seismic event. The Site Design Earthquake (SDE) is defined as, "the maximum predicted earthquake effect on the nuclear power plant, at a particular site, having an occurrence rate of 0.01 per year, based on historical records of actual earthquakes applicable to the site, and is expressed in the form of response spectra." The SDE is applied to the nuclear power plant structures which are to be seismically qualified to that level of design earthquake. The minimum ground-motion acceleration for the SDE is 0.03 g but is usually related to the seismic zone on which the National Building Code of Canada is based.

The DBE and SDE are arrived at independently, and thus bear no direct relationship to each other (i.e., no fixed ratio of maximum ground motions). The DBE is comparable to the Safe Shutdown Earthquake (SSE) as defined in the US. The SDE is comparable in level (not in application) to the Operating Basis Earthquake (OBE) in the US.

In the design of any given system or structure only one of the two seismic severity levels (DBE or SDE) is considered, except in the case of the containment system which is checked for both levels using different load factors, in order to determine which level governs. Currently the DBE governs containment design.

As is shown in Figure V.1 the entire PHTS of the CANDU reactor is qualified by design for the DBE; this includes the core and pressure tubes, which can stand earthquakes with ground acceleration of 0.5 g and higher.[21] The Canadian licensing criteria do, for that reason, not require consideration of a large break in the PHTS as a consequence of a seismic event; leaks due to an earthquake are, however, accommodated by the design. The Canadian licensing approach for CANDU reactors differs in this area from that in the US for LWRs, where licensing criteria do require consideration of a large-scale LOCA simultaneously with a seismic event of the severity level of the SSE. It should be noted, however, that the Canadian requirements concerning protection against seismic events are in complete agreement with the IAEA Codes of Practice and Safety Guides for Nuclear Power Plants, which is quite specific on the point that a system qualified for a seismic event of a certain severity level is not required to be assumed failed in a catastrophic manner following such an event. While the US safety approach is more conservative than that in Canada with respect to the assumption of the simultaneous occurrence of a maximum-size LOCA and the maximum-severity level earthquake, the Canadian safety approach is more conservative as regards the assumption of containment impairment.

Because the Canadian licensing criteria for CANDU reactors do not require consideration of a large-scale LOCA simultaneous with an earthquake, the ECIS is not required to be qualified for the DBE. The ECIS is, however, qualified for the SDE, so as to be able to continue to provide core-cooling capability if, during the recovery period following a postulated non-mechanistic large-scale LOCA, a seismic event were to occur of the severity level of the SDE (which, per definition, has a relatively "high" frequency of 10^{-2} per year).

In view of the above considerations it cannot be assured that the ECIS will function properly following a DBE. In case of a postulated leak in the PHTS due to a DBE, cooling of the core is to be provided by the Emergency Water Supply (EWS), powered by the Emergency Power Supply (EPS); both of these latter systems are fully qualified by design for the DBE (see Figure V.1).

The Canadian licensing criteria require the containment system to be qualified by design for the DBE and other natural phenomena including tornados, hurricanes, etc. Since Canadian licensing criteria for the CANDU reactor do not require consideration of a large-scale LOCA following a DBE but only a leak in the PHTS, the containment system is designed for loads due to the DBE combined with a coincident containment pressure up to that at the onset of the containment energy removal system-dousing system or vacuum building (∿9 psig in the case of a single-unit containment).

As regards failures of single components in safety systems, Canadian licensing procedures for CANDU reactors require meeting the same, or similar, criteria as those in force in the US for LWRs with respect to redundancy, diversity, separation, and independence. This holds particularly true for active components. An exception exists for some passive components such as low-pressure piping, where in some cases Canadian design criteria do not require redundancy for CANDU reactors; this is in particular the case for safety systems having diverse back-up systems (see discussion in Chapter VII).

An important characteristic of the Canadian safety design approach towards obtaining high reliability is that in many cases "redundancy in safety systems" is provided for protection against certain accident sequences, whereas the approach in the US is often to provide a single redundant system. As examples of this difference in design approach may serve: two sets of diesel-generators for CANDU reactors versus one set for LWRs; Service Water Supply, Auxiliary Feedwater Supply, and Emergency Water Supply for CANDU reactors (see Figure V.1) versus Service Water Supply and Auxiliary Feedwater Supply for LWRs; two diverse rapid shutdown systems, each capable of attaining cold reactor shutdown, for CANDU reactors versus one rapid shutdown system for LWRs; two cooling systems capable of preventing loss of coolable core configuration and core meltdown following a large-scale LOCA (ECIS and Moderator Cooling System) for CANDU reactors versus one cooling system (ECCS) for LWRs, etc.

C. Design Criteria Relative to Safety Analysis

Table V.7 gives a non-exhaustive matrix of design basis accidents of the single- and dual-failure type, considered in the Canadian licensing process for CANDU-PHW reactors. In the dual-failure category, each type of process-system failure is combined with failure of any one of the safety systems. In some instances these combinations result in trivial cases, not requiring analysis. In general, accidents in the dual-failure category are more restrictive as regards design requirements than those in the single-failure category; this obtains particularly for LOCAs combined with containment impairment, or for LOCAs combined with failure of the ECIS.

As mentioned in the foregoing, Canadian licensing regulations for CANDU reactors are in some areas more conservative than those in force in the US for LWRs, whereas in other areas the opposite is true. The former situation holds particularly true for the category of postulated dual-failure accidents: The US licensing process for LWRs does not require consideration of containment impairment or ECCS failure in conjunction with a LOCA.

Canadian licensing criteria for CANDU reactors allow to assume the correct performance of the ECIS for the analysis of a postulated LOCA combined with containment impairment. In this case the designer has to show by analysis that the radiological doses are consistent with the reference doses for dual-failure accidents. Similarly, for the analysis of a postulated LOCA combined with failure of ECIS, it is allowed to assume the correct performance of the containment system. In this case, a considerable fraction of the radioactive fission products may be released from the PHTS, and must be retained by the containment system. It should be mentioned in this connection that failure of the ECIS in a CANDU reactor does not result in core meltdown, since the moderator constitutes a large dispersed heat sink, with a long term heat removal capability equal to $\sim$5% of nominal power.

The spectrum of loss-of-coolant accidents required to be considered in the Canadian licensing procedure for CANDU reactors is similar to that required in the US for LWRs, and covers the full spectrum of failures of the PHTS up to and including the so-called "100% break" of the largest-diameter piping. Failures of sufficient magnitude and duration in the PHTS will result in blowdown of part of (Pickering), or the entire (Bruce) primary coolant system, depending on whether the PHTS is divided into two or more independent systems, or whether it consists of a single system. Such LOCAs would require correct performance of the ECIS in order to limit damage to the fuel.

The so-called "100% break", postulated in the Canadian licensing process, is defined as having a cross-sectional flow area equal to twice the value of the cross-sectional flow area of the affected header or pipe, and is assumed to occur instantaneously; it differs slightly from the so-called "double-ended rupture", postulated in US LOCA analyses, in that in the Canadian case the rupture is postulated to occur on one side of the header or pipe, without resulting in a complete circumferential rupture and subsequent off-set as for the US postulated break. The outcome of the analysis of postulated LOCAs for the CANDU-PHW reactor has been found to be relatively insensitive to minor differences in the initial assumptions, such as the difference between the 100% break and the double-ended break (see further under Chapter VI).

The Canadian licensing criteria imposed for CANDU reactors in the evaluation of postulated LOCAs are slightly different from those currently in force in the US for LWRs. The current US interim licensing criteria for the ECCS, having as principal objectives maintaining coolable core configuration and keeping the energy release due to metal-water reaction at a negligible level, are as follows:

(a) the maximum fuel cladding temperature shall not exceed 1200°C (2200°F), and

(b) the calculated oxidation of the cladding shall nowhere exceed 17% of the total cladding thickness before oxidation.

The before-mentioned temperature limit of 1200°C is imposed solely on the basis of oxygen embrittlement; melting of the cladding and/or fuel, energy release from the zirconium-steam reaction, and damage by eutectic formation are not a concern at this temperature.

The Canadian licensing position, supported by a considerable body of experimental data produced at the Whiteshell and Chalk River Nuclear Research Establishments as well as elsewhere[32,33,34] is that a strict temperature limit (1200°C) as part of the oxygen embrittlement criteria (instead of a temperature-time relationship), is extremely conservative. The current Canadian design criteria for ECIS performance therefore do not include a strict limitation to <1200°C, but do require that the oxygen concentration be less than 0.7% over at least half the cladding thickness. Figure V.2 gives an example of the time-temperature relationship for oxygen embrittlement of zircaloy, showing the difference between Canadian and US criteria. It would seem that this Canadian criterion, though perhaps somewhat less conservative than its US counterpart, does meet the intent of the US ECCS licensing criteria, namely avoidance of excessive zircaloy-cladding embrittlement.

The US regulatory guidelines with respect to the release of radioactive material from the fuel and the containment subsequent to a LOCA are as follows (US NRC Regulatory Guide 1.3):

(a) 25% of the equilibrium radioactive iodine inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the primary reactor containment. Of this 25%, a fraction of 91% is assumed to be in the form of elemental iodine, a fraction of 5% is assumed to be in the form of particulate iodine, and a fraction of 4% is assumed to be in the form of organic iodires.

(b) 100% of the equilibrium radioactive noble gas inventory developed from maximum full power operation of the core should be assumed to be immediately available for leakage from the reactor containment.

The Canadian position is that the assumption of instantaneous release of 25% of all radioactive iodine and 100% of all radioactive noble gas is overly conservative, since (apart from the prompt release of a portion of the free fission products in the fuel pin gaps) the fission product release from the fuel matrix is predominantly governed by the fuel temperatures attained, and by the fuel temperature distributions in the fuel pins and in the core as a whole. For different radioactive fission products (I-131, Ru-106, Cs-137, Sr-89 Sr-90, Xe, Kr) at the same fuel temperature the release fraction is different. The Canadian licensing practice, with respect to release of radioactive material from the fuel pins following a LOCA therefore requires:

(i) calculation of cladding and fuel temperature transients, subsequent to coolant blowdown and initiation of cooling by the ECIS,

(ii) determination of the failure fraction of fuel pin cladding (criteria for cladding failure: (1) 5% uniform strain, or (2) excessive oxygen embrittlement due to oxygen concentration > 0.7% over at least half the cladding thickness).

(iii) calculation of the quantity of fission products released, from fuel pins with failed claddings, on the basis of the calculated spatial temperature transients in the fuel attained during blowdown and subsequent cooling by the ECIS. It is conservatively assumed that the reactor has been operated continuously at 100% prior to the LOCA.

It would seem that the Canadian licensing practice with respect to release of radioactive material from the fuel into the containment is fully justifiable on technical grounds.

TABLE V.1 Division Between Process Systems and Safety Systems

Process Systems	Safety Systems
- Primary Heat Transport System	- Shutdown System 1 (SDS-1)
- Reactor Control System(s)	- Shutdown System 2 (SDS-2)
- Electrical Systems	- Emergency Coolant Injection System
- Turbine	- Containment System
Etc.	

TABLE V.2 Reference Dose Limits in Canada

Plant Condition	Maximum Frequency Allowed	Meteorology to be used in Calculation	Maximum Individual Dose Limits	Maximum Total Population Dose Limits
Normal Operation		Weighted according to effect, i.e. frequency times dose for unit release	0.5 rem/yr whole body 3 rem/yr to thyroid (1)	10^4 man-rem/yr 10^4 man-rem/yr to thyroid
Serious Process Equipment Failure	1 per 3 years	Either worst weather existing at most 10% of time or Pasquill F condition if local data incomplete	0.5 rem whole body 3 rem to thyroid	10^4 man-rem whole body 10^4 man-rem to thyroid
Process Equipment Failure plus Failure of any Safety System	1 per 3×10^3 years	Either worst weather existing at most 10% of time or Pasquill F condition if local data incomplete	25 rem whole body 250 rem to thyroid (2)	10^6 man-rem whole body 10^6 man-rem to thyroid

(1) For other organs use 1/10 ICRP occupational values
(2) For other organs use 5 times ICRP annual occupational dose (tentatively)

TABLE V.3 Dose Limits in US

US-NRC	EPA
Permissible Levels of Radiation in unrestricted areas (10CFR20.105) (a) For average radiation levels and anticipated occupancy: 0.5 rem/yr whole body (b) Radiation levels causing dose of: 2 mrem in one hour or 100 mrem in 7 consecutive days (c) As low as reasonably achievable from effluent releases, 5 mrem/yr target	Annual dose equivalent to any member of the public shall not exceed: Planned discharges (normal operations) 0.025 rem/yr whole body 0.075 rem/yr thyroid 0.025 rem/yr any other organ Per Gigawatt-year 50,000 Ci Kr-85 5 m Ci I-129 0.5 m Ci Pu-239 and other alpha-emitting transuranics

Reactor Site Criteria (for major accidents) (10CFR100.11)

	rem Whole Body	rem Thyroid
(a) Site boundary in 1st 2 hours	25	300
(b) Low Population Zone during cloud passage.	25	300

TABLE V.4 Canadian General Power Reactor Safety Criteria & Principles

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1. Design and construction of all components, systems and structures essential to or associated with the reactor shall follow the best applicable code, standard or practice and be confirmed by a system of independent audit.

2. The quality and nature of the process systems essential to the reactor shall be such that the total of all serious failures shall not exceed 1 per 3 years. A serious failure is one that in the absence of protective action would lead to serious fuel failure.

3. Safety systems shall be physically and functionally separate from the process systems and from each other.

4. Each safety system shall be readily testable, as a system, and shall be tested at a frequency to demonstrate that its (time) unreliability is less than 10^{-3}.

5. Radioactive effluents due to normal operation, including process failures other than serious failures (see #2 above), shall be such that the dose to any individual member of the public affected by the effluents, from all sources, shall not exceed 1/10 of the allowable dose to Atomic Energy Workers and the total dose to the population shall not exceed 10^4 man-rem/year.

6. The effectiveness of the safety systems shall be such that for any serious process failure the exposure of any individual of the population shall not exceed 500 mrem and of the population at risk, 10^4 man-rem.

7. For any postulated combination of a (single) process failure and failure of a safety system, the predicted dose to any individual shall not exceed (i) 25 rem whole body, (ii) 250 rem, thyroid, and to the population, 10^6 man-rem.

8. In computing doses in 6 and 7, the following assumptions shall be made unless otherwise agreed to:

   (i) meteorological dispersion that is equivalent to Pasquil category F as modified by Bryant.

   (ii) conversion factors as given by Beattie.

TABLE V.5 Division of Systems in Two Groups to Protect Against Common-Mode Events

Group 1	Group 2
- Control Systems	- Shutdown System 2
- Safety Shutdown System 1	- Containment System
- ECIS	- Emergency Power Supply
- All Process Systems (except auxiliary moderator cooling system)	- Emergency Water Supply
	- Emergency Instrumentation for Plant-Status Monitoring

TABLE V.6: Design Classification of Main Systems of the CANDU-PHW Reactor

System	ASME		Seismic Qualification Level	
	Section	Class	System	Power Supply
(1) PHTS				
- Fuel Channel Assembly	III	1	DBE	
- Feeders	III	1	DBE	
- Headers	III	1	DBE	
- Steam Generators (Primary Side)	III	1	DBE	
- Pressurizer	III	1	DBE	
- Pumps	III	1	DBE	
(2) ECIS	III	1	DBE	Not
(3) Moderator Cooling System	III	2	DBE	Not
(4) Calandria & Endshields	III	2	DBE	
(5) Shutdown Cooling System	III	1	DBE	Not
(6) Coolant Make-up System	III	1 & 3	Partially to DBE	Not
(7) Feedwater Supply System	Non-Nuclear		Partially to DBE	Not
(8) Auxiliary Feedwater System	Non-Nuclear		Partially to DBE	Not
(9) Service Water System	Non-Nuclear		Partially to DBE	Not

TABLE V.7 Accident Matrix */

Process-System Failure	Single-Failure Accidents	Dual-Failure Accidents SDS-1 Failure	SDS-2 Failure	ECIS Failure	Containment Failure
- Loss-of-Regulation	X	X	X	--	--
- Loss-of-Coolant	X	X	X	X	X
- Loss-of-(Primary)-Heat-Sink **/	X	X	X	--	--

*/ Postulated accidents indicated by X require analysis, whereas postulated accidents indicated by -- are trivial cases

**/Postulated Loss-of-Heat-Sink accidents require assessment of alternative heat sinks.

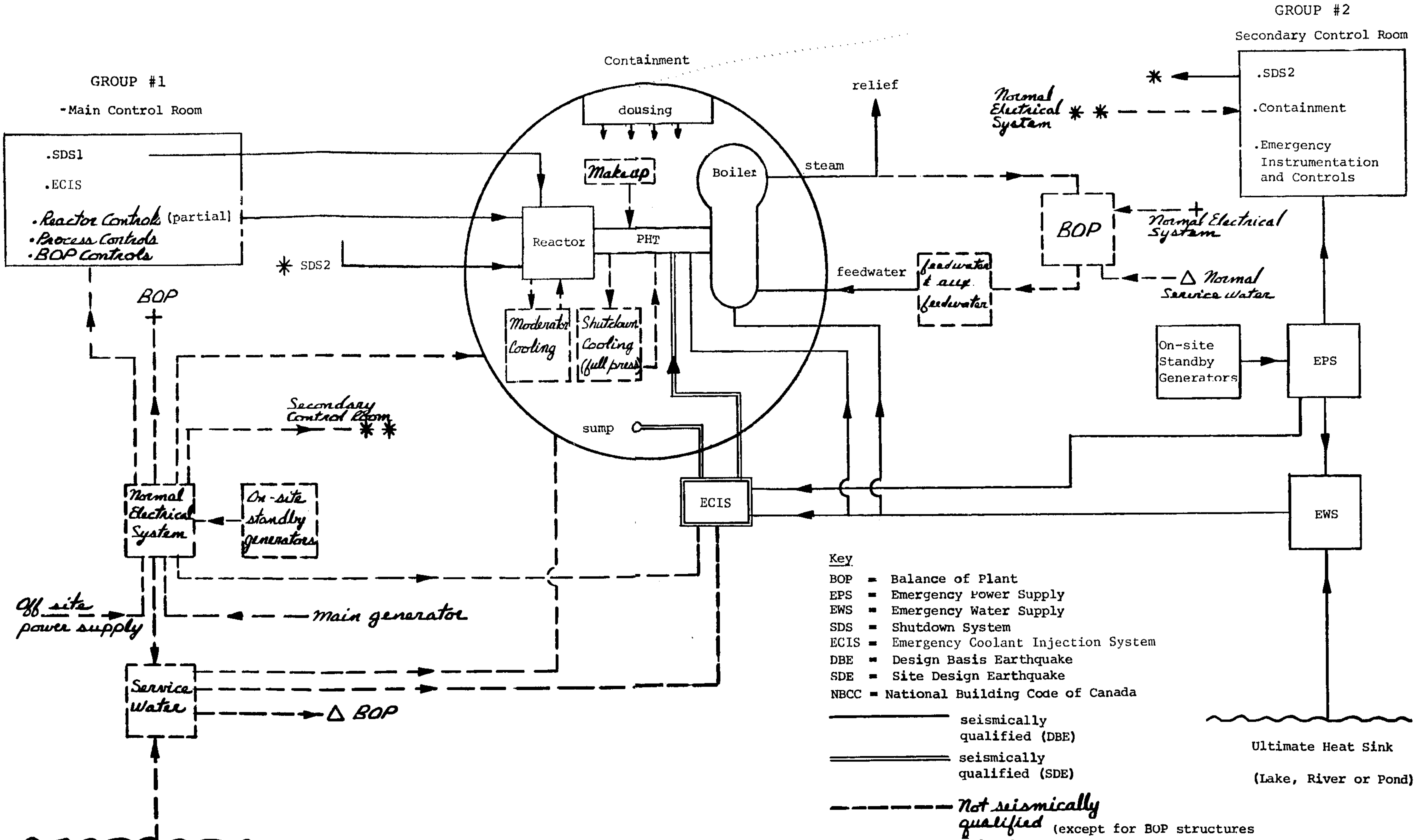

Fig. V.1 Seismic Qualifications and Systems Separation for a 600 MWe Station on an Ocean Site

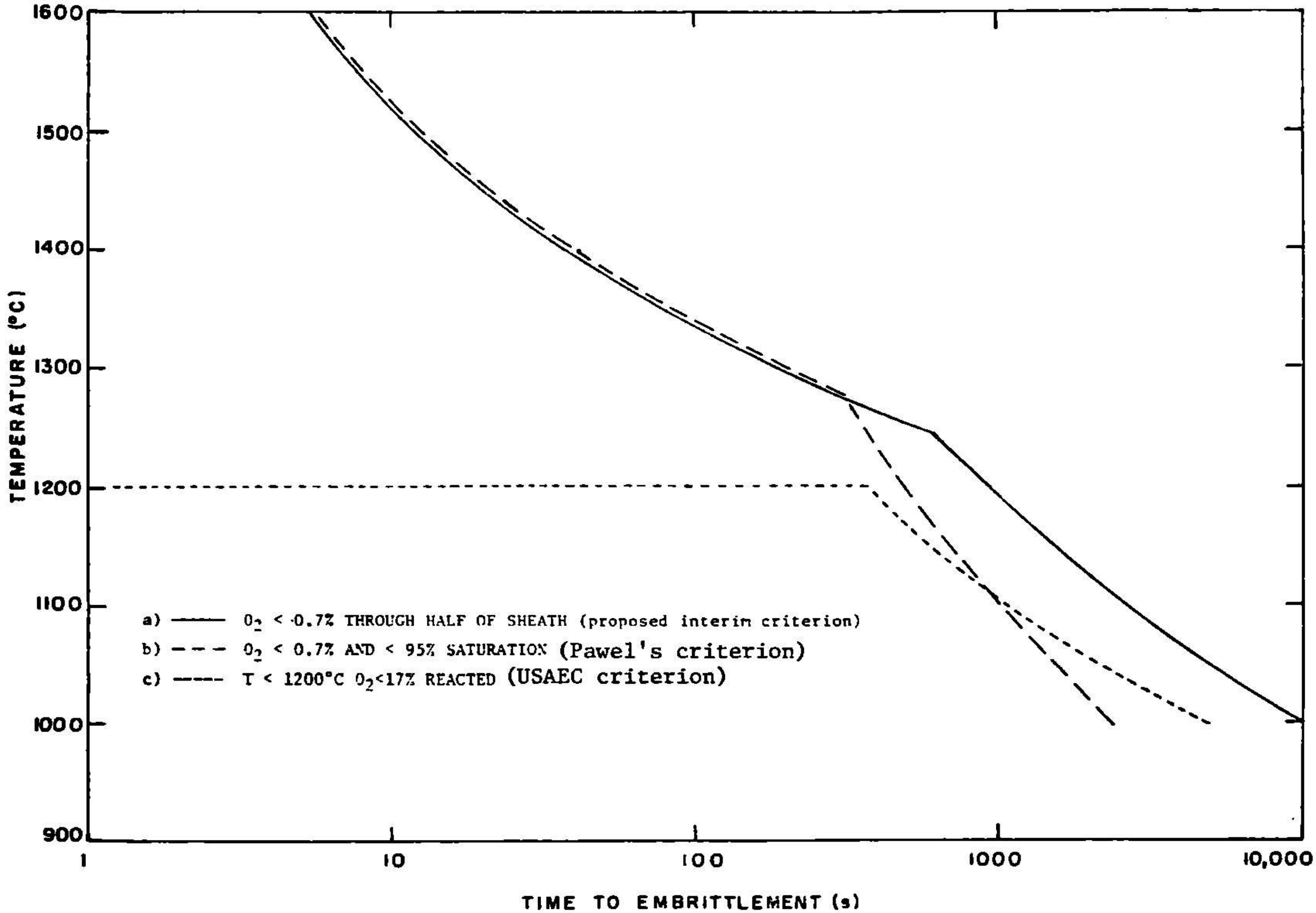

Fig. V.2 An Example of the Time-Temperature Relationship for Embrittlement as Determined by Embrittlement Criteria for Isothermal Oxidations of 0.42 mm Thick Cladding

VI. CANDU-PHW Safety Analysis

A. General Aspects

The spectrum of postulated accident sequences, required to be analyzed for CANDU-PHW reactors as part of the Canadian regulatory process, has many similarities with, and is of equal breadth as, that required for LWRs in the US. The objective of this Chapter is not so much to present detailed analyses, but rather to describe the principal characteristics of the CANDU-PHW safety analysis, and to identify essential differences with LWR safety analysis, where appropriate. Such differences may be due to a number of causes, among which (1) intrinsic differences in the design characteristics of CANDU reactors and LWRs, (2) differences in safety approach, and (3) differences in the criteria applied.

One intrinsic characteristic of the CANDU-PHW reactor, which it shares with LWRs and which has not been mentioned in earlier Chapters, is that its fuel is arranged in the configuration of maximum reactivity; any displacement of the fuel in the core will lead to a reduction of reactivity.

The principal categories of postulated initiating events leading to various types of postulated accident sequences, which are considered in the CANDU safety analysis, are the following:

(1) Reactivity Accidents or Loss-of-Regulation Accidents (LORAs)

(a) at nominal power

(b) during startup: - from cold conditions

- from hot conditions

(2) Loss-of-Flow Accidents (LOFAs)

(a) pump coastdown: - single pump

- multiple pumps

(b) pump seizure or pump shaft break (single pump only)

B. Loss-of-Regulation Accidents (LORAs)

As has been observed earlier, the total control reactivity needed in a CANDU-PHW reactor is relatively small, because of the application of on-load refueling. The total reactivity that potentially can be involved in a LORA is therefore limited.

The postulated initiating event that sets an upper bound as regards the rate of reactivity insertion (the design basis LORA) is the simultaneous uncontrolled withdrawal of all neutron poison devices at their maximum speed. There is no credible (mechanistic) set of circumstances that could result in a higher rate of reactivity insertion through loss of regulation; one consideration in this connection is that all neutron poison devices are installed within the low-pressure moderator region, ruling out the possibility of control rod ejection. It needs to be emphasized that the occurrence of an event in which all neutron poison devices are withdrawn at their maximum speed is extremely improbable, because of the redundancy and separation (e.g., two separate computers) provided for in the CANDU-PHW control system, as well as the presence of numerous built-in interlocks and self-checking features.

Protection against LORAs is provided by eight different reactor trips, of which four on each of the reactor shutdown systems (SDS1 and SDS2). These reactor trips are (1) reactor trip for startup protection in the source-range power level, (2) reactor trip with setpoint at $\sim$1% of nominal power for protection in the intermediate power level range, (3) reactor trip at $\sim$110% of nominal power for protection at full power, and (4) reactor period (log-rate) trip for protection over most of the entire range. Furthermore, each reactor shutdown system (SDS1 and SDS2) is provided with an extensive regional overpower protection system, which in case of slow transients (when the rate trips are not invoked) provide regional core protection. Both reactor shutdown systems

(SDS1 and SDS2) have each separately complete capability to shut the reactor down from full power operation to cold conditions.

The analysis of LORAs for CANDU-PHW reactors does not differ to any substantial degree from that for LWRs. It should be noted, however, that the mean neutron life time in CANDU reactors is, as mentioned earlier, a factor $\sim$30 larger than in LWRs. This fact, combined with the presence of two independent reactor shutdown systems and the relatively small control reactivity needed in CANDU reactors, results on the whole in a small potential for severe power excursions in CANDU-reactors.

Of primary concern in the analysis of LORAs is the total energy accumulated in the fuel pins during the short power excursion. Analyses made for the design basis LORA, in which all control rods are withdrawn simultaneously at their maximum speed, indicate that, even for the most conservative assumptions (e.g., maximum accumulative instrument error with respect to initial power level, etc.), the critical heat flux (CHF) is not reached and no core damage will occur.

C. Loss-of-Flow Accidents (LOFAs)

Loss-of-flow accidents, due to electrical or mechanical failure affecting one or more PHTS pumps, or due to loss of power to all pumps, is analyzed for CANDU-PHW reactors in a similar manner as for LWRs. As for pressurized water reactors (PWRs), the PHTS pumps of CANDU-PHW reactors are provided with high rotational inertia resulting in a sufficiently long coastdown in case of loss of power to avoid any fuel damage for all types of postulated LOFAs. The fact that the PHTS of the CANDU-PHW reactor is subdivided in separate loops, each having a figure-of-eight configuration (see Figure II.2) with two pumps in series, provides an additional safety margin against failures affecting a single pump. Critical heat fluxes are calculated using a correlation similar to those used for LWRs. This AECL correlation is based on an extensive body of experimental data, generated in Canada and elsewhere.

Instantaneous seizure of a single pump constitutes the most severe LOFA, having a very low probability of occurrence. It results in a rapid reduction of the flow in the affected loop to about 60% of its nominal value, and is followed immediately by reactor shutdown initiated by two signals (generated by redundant sets of independent and diverse sensors), resulting in a rapid reduction of the reactor power to decay heat level. During the short transient, in which the stored heat in the fuel pins is removed, transition boiling conditions may be reached for a few seconds at the surface of the fuel in some limited core regions served by the affected loop. Analyses indicate, however, that even for this postulated severe type of LOFA, no fuel failure is expected to occur. It should be emphasized again that instantaneous seizure of a PHTS pump is an event of extremely low probability in view of the continuous monitoring and in-service inspection program to which the PHTS pumps and their bearings are subjected. Any tendency to seizure of the bearings is preceded by a period of increased vibration and power consumption of the affected pump-motor set, which would provide ample warning and adequate time for an orderly shutdown of the reactor prior to sudden and complete seizure.

For all of the other types of postulated LOFAs, the consequences are less severe than for the case of postulated instantaneous seizure of a single pump, because of the presence of the high rotational inertia. As an example, for the case of a postulated simultaneous coastdown of all pumps, followed by reactor shutdown, analyses indicate that at no time will critical heat flux conditions be reached anywhere in the core.

D. Loss-of-Coolant Accidents (LOCAs)

Figure II.1 gives the layout and principal characteristics of a single loop of the PHTS of the CANDU-PHW. As mentioned earlier, the loop is arranged in a figure-of-eight configuration with one reactor inlet header (RIH) and one reactor outlet header (ROH) on each side of the core (four headers in total per loop). Each PHTS loop is provided with two pumps and two boilers (steam generators). As shown in Figure II.4, the feeders of the reactor power channels are connected to the headers. The ECIS has injection points in every inlet and outlet header. The moderator, which, at full power, receives $\sim$ 5% of the nominal reactor power (primarily due to neutron slowdown and gamma heating) is cooled by means of a separate cooling system with redundant pumps (see Figures II.1 and II.18). Furthermore, the entire moderator region is surrounded by a light water shield (see Figures II.10, II.11, II.12, and II.13), which again has a separate cooling system with heat removal capability equal to 0.3% of nominal reactor power.

Loss of piping integrity, resulting in a LOCA, can be postulated to occur with various degrees of severity (various break or crack sizes) and in different locations of the PHTS. The probability of the occurrence of a sudden large-size break is extremely small, because of the "leak-before-break" characteristic of PHTS components, and because of the extensive leak detection systems and inservice surveillance programs. Still, breaks of all sizes are evaluated as part of the CANDU-PHW safety analysis. Among the breaks postulated to take place outside the core region are three break types that are limiting, namely

1) 100% break of the pump suction piping, which results in the highest coolant discharge rate into the containment and thus in the highest containment pressure peak;

2) 100% break of the inlet header, which results in the highest initial coolant voiding rate in the core region, and thus in the highest rate of positive reactivity insertion; and

3) 35% break of the inlet header, which results in the highest fuel cladding temperatures.

For reasons of convenience LOCAs are subdivided into three phases, namely:

a) the blowdown phase, during which the PHTS pressure drops from its normal operating level (11 MPa) to the ECIS injection pressure,

b) the rewetting and refilling phase, during which the ECIS injects coolant into the PHTS and causes the fuel elements to be rewetted and the PHTS to be refilled, and

c) the post-accident recovery phase, during which the ECIS provides long-term cooling.

The correlations that are used in the analysis of LOCAs are the following:

i) Blowdown: Fauske-Henry correlation;

ii) Frictional Pressure Drop: Martinelli-Nelson with Collier-Jones flow correction;

iii) Heat Transfer:

- convective heat transfer: Dittus-Boelter correlation,
- subcooled boiling: Thom correlation,
- nucleate boiling: Schrock-Grossman correlation,
- film boiling: Groeneveld correlation;

iv) Critical Heat Flux: AECL correlation (which depends on the fuel bundle configuration);

v) Metal-water reaction: Baker-Just and GE correlations.

Among the breaks that are postulated to take place outside the core, those that occur at the inlet side of the core region are more severe than those on the outlet side, because (1) the coolant temperature is about 45°C lower and the coolant pressure is about 1.25 MPa (180 psi) higher at the Reactor Inlet Header (RIH) than at the Reactor Outlet Header (ROH), resulting for the same break size in higher discharge rates of the coolant for a rupture at the inlet than at the outlet side of the core, and (2) for an inlet break, flow stagnation and flow reversal will take place in an early phase of the blowdown, resulting in deterioration of the heat removal process in the affected power channels of the core when the heat flux at the fuel surface is still quite high. On the other hand, for a break at the core outlet, the flow rate through the core initially tends to increase, so that initially heat removal capability is maintained and deterioration of heat removal capability occurs in a later phase of the blowdown process, when the power level and heat flux into the coolant has decreased already substantially due to reactor shutdown.

Each loop of the PHTS has design provisions for its isolation, following a break in the other loop, so that it is highly improbable that, in case of a LOCA, both loops would be affected simultaneously (see Figure II.17). Notwithstanding this, it is conservatively assumed for the design of the containment system that during a LOCA the intact loop will also be affected, blowing down its coolant through the interconnecting piping at a slower rate than the loop with the break.

In order to lower rapidly the pressure in the PHTS following a LOCA, thus allowing early introduction of ECIS coolant into the PHTS, a so-called "rapid safety cooldown" is initiated (by signals detecting and verifying the occurrence of a LOCA), in which the coolant on the secondary side of the steam generators is blown off to atmosphere. This procedure is of particular importance for postulated small-size breaks when the pressure in the PHTS tends to remain high for a prolonged time period; it is, however, not required for postulated large-size breaks of the headers.

Adequate heat removal capability (also for the long-term) for the core and the containment is provided through incorporation of appropriate design features (see Figures VI.1 and VI.2), even for the case that the LOCA were to be followed during the recovery period by a seismic event of the severity level of the SDE; for this latter case reliable backup heat removal capability is provided by the EWS powered by the EPS (see Figure V.1).

Figures VI.3 through VI.9 present some typical results of CANDU-PHW LOCA analysis.

Because of their specific configuration, each of the loops of the PHTS comprises two separate core sections in which the coolant flows in opposite directions. These two core sections will be affected differently by a LOCA, because of their different distances to the break: The core section which is located upstream of the break will lose its coolant slower than the downstream core section (see Figure II.2). It is important to be able to calculate the coolant hold-up in the core as a function of time following a loss of piping integrity, because of the positive void-reactivity effect. Figure VI.3 gives a typical representation of the coolant density as a function of time in the upstream and downstream core sections.

Immediately following the LOCA, the reactor power will rise slightly due to reactivity insertion caused by coolant voiding in the core region. Figure VI.4 presents the reactor power as a function of time for various break sizes in the Reactor Inlet Header (RIH).

Figure VI.5 gives the various reactivity contributions as a function of time for a 100% RIH break for the case of a fresh fuel core. Figure VI.6

gives temperatures of the fuel, cladding, coolant, and pressure tube wall, as a function of time.

Figure VI.7 gives the total energy released into the containment as a function of time, also for the case that the isolating valves between the two loops fail to close. This latter case is used as the Design Basis Accident for the containment. Figure VI.8 gives the containment pressure in a single-unit containment system as a function of time for the initial phase of the blowdown. Figure VI.9 presents containment pressure versus time over the entire transient, and shows the effect of the dousing system.

The maximum cladding temperature, which is encountered in the hotspot of the core for a 35% Reactor Inlet Header (RIH) break is found to be about 1200°C; this is at the limit of allowable cladding temperatures (<2200°F) as specified by US-NRC 10-CFR-50, Appendix K (as mentioned in Chapter V, Section C, Canadian requirements are more flexible on this point, allowing higher cladding temperatures, if the duration of the high cladding temperature remains within certain time limits, see Figure V.2).

For postulated LOCAs of the single-failure category due to breaks in the PHTS outside the core region, the safety analysis of the CANDU-PHW reactor shows that (1) the rate of reactivity insertion and the reactivity depth of each of the reactor shutdown systems (SDS1 and SDS2), taken independently, is adequate to overcome the positive reactivity transient due to voiding of the PHTS, and to shut the reactor down to cold conditions, (2) cooling of the fuel by means of the ECCS can be established for all break sizes while maintaining coolable fuel configuration in all regions of the core, (3) the pressure transient within the containment building does not exceed the design value, (4) radiation doses for single individuals and the population at large are consistent with the reference dose limits established by the AECB for accidents of the single-failure category.

For postulated LOCAs of the dual-failure category due to breaks in the PHTS outside the core region, the safety analysis depends on the particular safety system which is postulated to be unavailable. Since there are two fully independent reactor shutdown systems (SDS1 and SDS2), each having the capability to shut the reactor down to cold conditions, unavailability of any one of the shutdown systems will leave the accident sequence within the envelope of single-failure accidents. However, unavailability of the ECIS or impairment of the containment system, will, of course, affect the accident sequence considerably.

For LOCAs combined with unavailability of the ECIS, the safety analysis of CANDU-PHW reactors shows that (1) the depth and rate of each of the reactor shutdown systems (SDS1 and SDS2), taken independently, is adequate, (2) adequate heat removal capability is provided by the moderator cooling system (having redundant pumps) and the moderator heat capacity, backed up by the heat capacity of the light water shield, so that a coolable configuration is maintained in all core regions without fuel melting, even though it is conservatively assumed that clad coolant interaction will take place, thus providing an additional heat source, (3) the pressure transient within the containment building does not exceed the design value, taking into account the additional heat source due to cladding-coolant interaction, (4) radioactive material released from the fuel and the PHTS remains largely contained within the containment system; radiation doses for single individuals and the population at large are consistent with the reference dose limits established by the AECB for accidents of the dual-failure category.

For LOCAs combined with containment impairment, the safety analysis of

CANDU-PHW reactor shows that (1) the depth and rate of each of the reactor shutdown systems is adequate, (2) cooling of the fuel by means of the ECIS can be established for all break sizes while maintaining a coolable configuration in all regions of the core, (3) radiation doses for single individuals and the population at large are consistent with the reference dose limits established by AECB for accidents of the dual-failure category.

A LOCA within the core region, due to loss of integrity of a pressure tube, belongs to a category of postulated accidents, which pertains to pressure tube reactors, and which does not need to be considered for LWRs. As has been pointed out in Chapter III, the probability for the occurrence of a sudden large size rupture of a pressure tube is extremely low. Still, such a rupture is considered as part of the safety analysis of the CANDU-PHW reactor, showing the following results: (1) the probability of any failure propagation to other pressure tubes is extremely low, even if the pressure tubes are embrittled due to exposure to the neutron flux, (2) there may occur some localized denting of adjacent calandria tubes in the immediate vicinity of the pressure tube rupture, (3) some calandria tubes will be temporarily flattened, remaining however in the elastic region, (4) the rupture diaphragms of the calandria may blow off, (5) the integrity of the calandria is maintained (even for the extremely improbable case of postulated simultaneous rupture of a number of pressure tubes), (6) shutdown capability of the reactor by means of solid rods is amply maintained (even if it conservatively assumed that some safety rods in the immediate vicinity of the rupture location do not enter), (7) shutdown capability of the reactor by means of the gadolinium injection system (SDS2) is fully maintained.

The above safety analyses are based to a large extent on experimental data, generated in Canada and elsewhere (Italy, Japan, U.K., etc.).

E. Loss-of-Heatsink-Accidents (LOHA)

Under normal operating conditions heat generated in the reactor core is transfered in the steam generator and is in part rejected in the condensor.

After shutdown of the reactor under normal conditions, the decay heat is initially removed through the steam generators and the condensors, and subsequently through the Shutdown Cooling System, which has a heat removal capability of ∿ 1.2% of nominal reactor power, if operated at reduced primary coolant temperatures. Under accident conditions the Shutdown Cooling System can be connected into the PHTS at full operating pressure (∿ 10 MPa = 1450 psia*; see Figure V.1), at which time it has a heat removal capability of ∿ 7% of nominal reactor power.

Loss of feedwater supply constitutes one of the most probable causes for a LOHA. To protect against loss of feedwater, the CANDU-PHW is provided with an Auxiliary Feedwater System. Furthermore, as noted earlier, after reactor shutdown the Shutdown Cooling System can be used to remove the full decay heat, even if the PHTS is not yet depressurized. The Main Feedwater System, and the Auxiliary Feedwater System, are not qualified for seismic events of the severity level of the DBE or the SDE. These two latter systems are therefore backed up by the EWS, which is qualified for seismic events of the severity level of the DBE, and is powered from the EPS. The EWS is physically separated from the main reactor building and can be controlled from the auxiliary control room, so as to provide additional protection against events which could cause common-mode failures (tornados, airplane crashes, etc.). After the reactor has been shut down and partially depressurized, the EWS can

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* The Residual Heat Removal System found in LWRs, generally does not have the capability of being connected into the PHTS at full operating pressure, its maximum pressure usually being ∿ 4.14 MPa (∿ 600 psia).

also be connected directly to the PHTS. The EWS can also be connected to the secondary side of the steam generators after depressurization. The maximum delivery pressure of the EWS is 0.69 MPa (100 psia).

In case of a large-size steam line break, heat removal capability via the condensors is lost. For this kind of postulated accident the reactor is shut down and partially depressurized on the primary side of the PHTS by the blowdown on the secondary side. Long-term decay heat removal can be provided by the Shutdown Cooling System, or by the EWS powered by the EPS (see Figure V.1). It should be noted in this connection that rapid depressurization on the secondary side does not for CANDU-PHW reactors result in a significant reactivity transient (as is the case for PWRs), and furthermore that each of the two shutdown systems (SDS-1 and SDS-2) of CANDU-PHW reactors controls sufficient reactivity to enable reactor shutdown to cold conditions (for PWRs, the reactor shutdown system has a reactivity worth sufficient only for reactor shutdown to hot-standby conditions).

To ensure PHTS integrity following a LOHA, Canadian practice requires demonstration by analysis that no steam generator tube failures will result from postulated large-size ruptures affecting either the feedwater supplies or the steam lines.

It is concluded, that CANDU-PHW reactors are adequately protected against the consequences of LOHAs, including having adequate provisions to guarantee long-term heat removal, even for very conservatively postulated accident sequences.

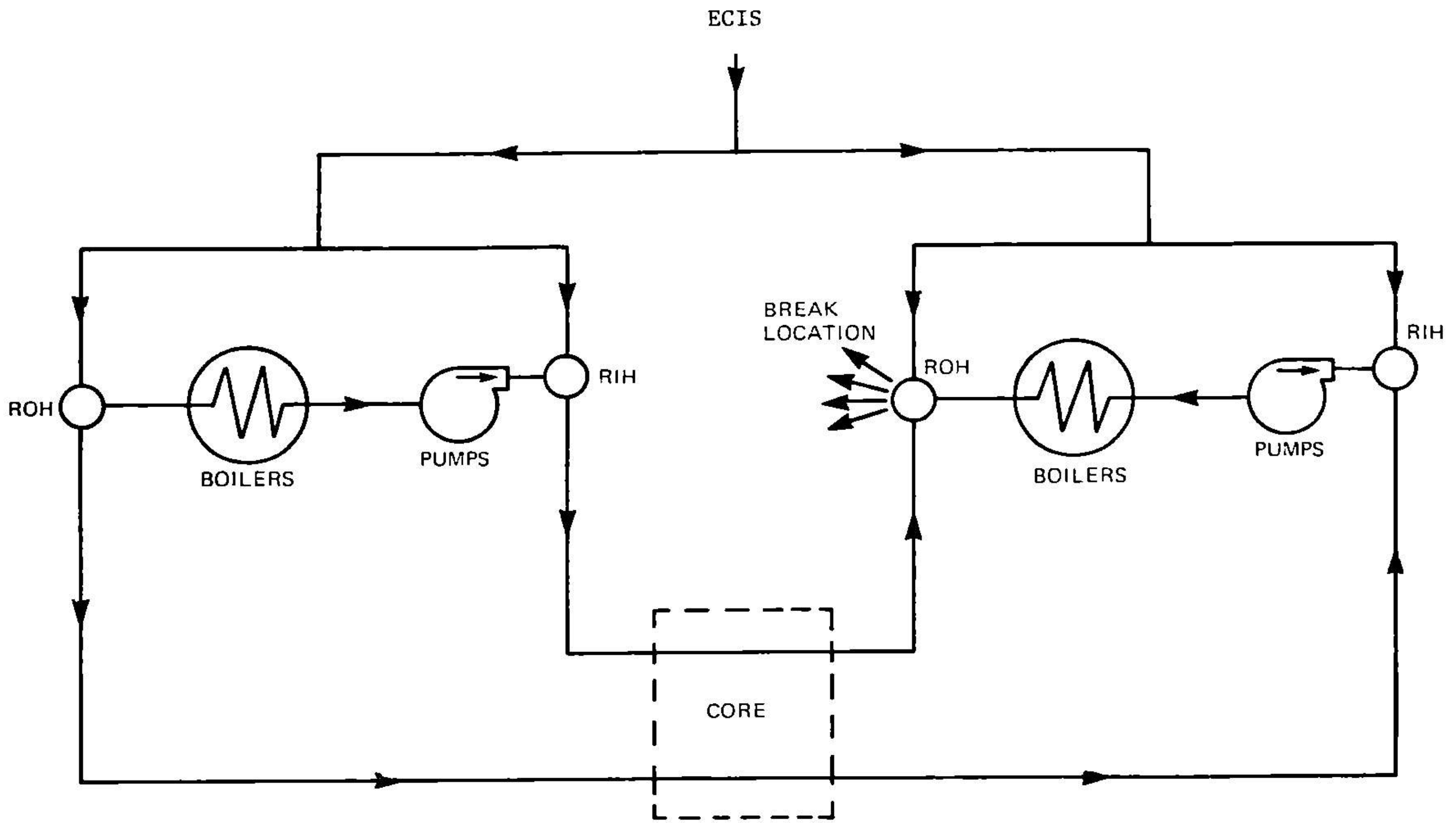

Fig. VI.1 Once-Through Flow Typical of Larger Breaks

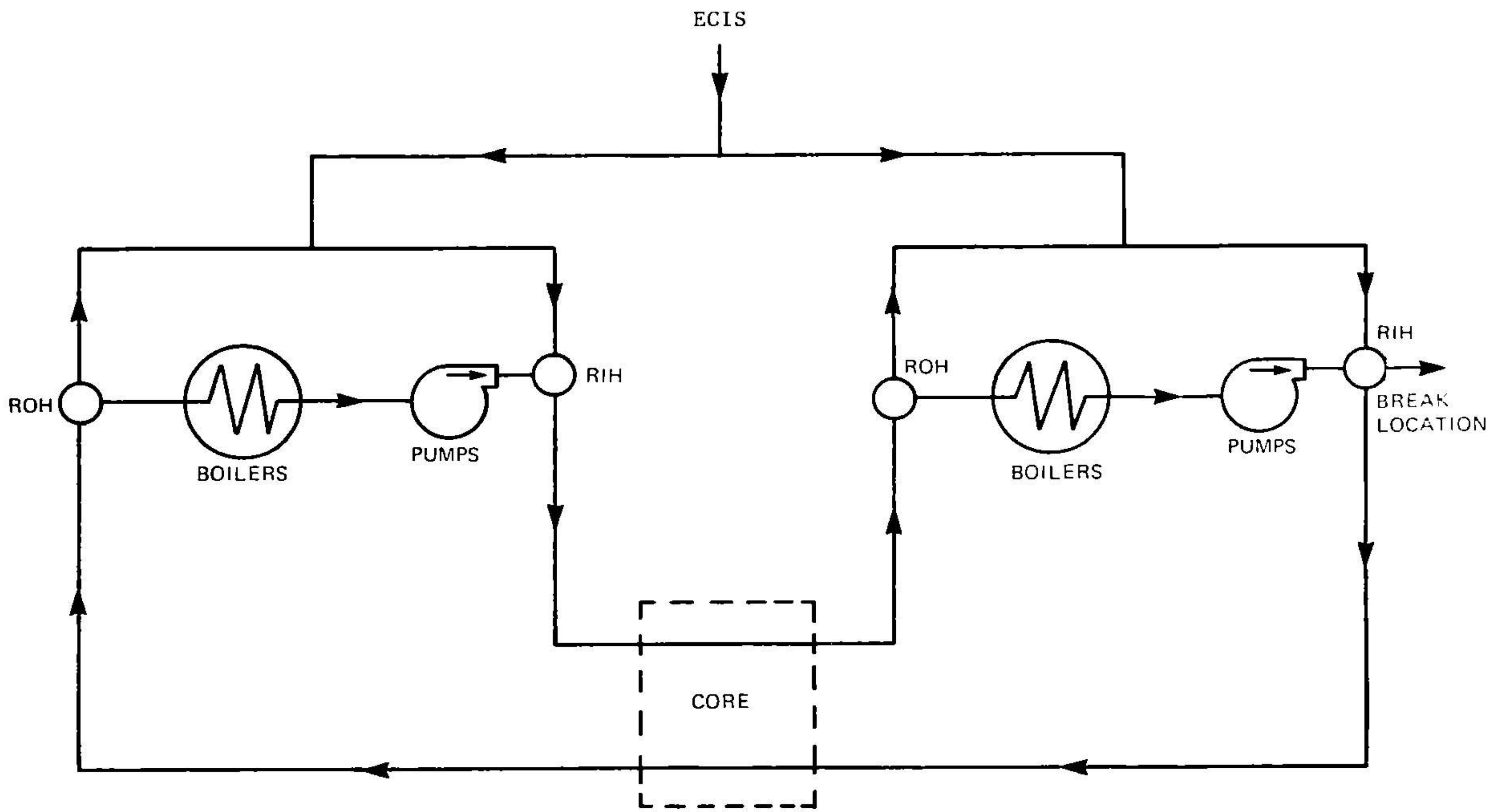

Fig. VI.2 Normal Re-Circulation Flow Typical of Smaller Breaks

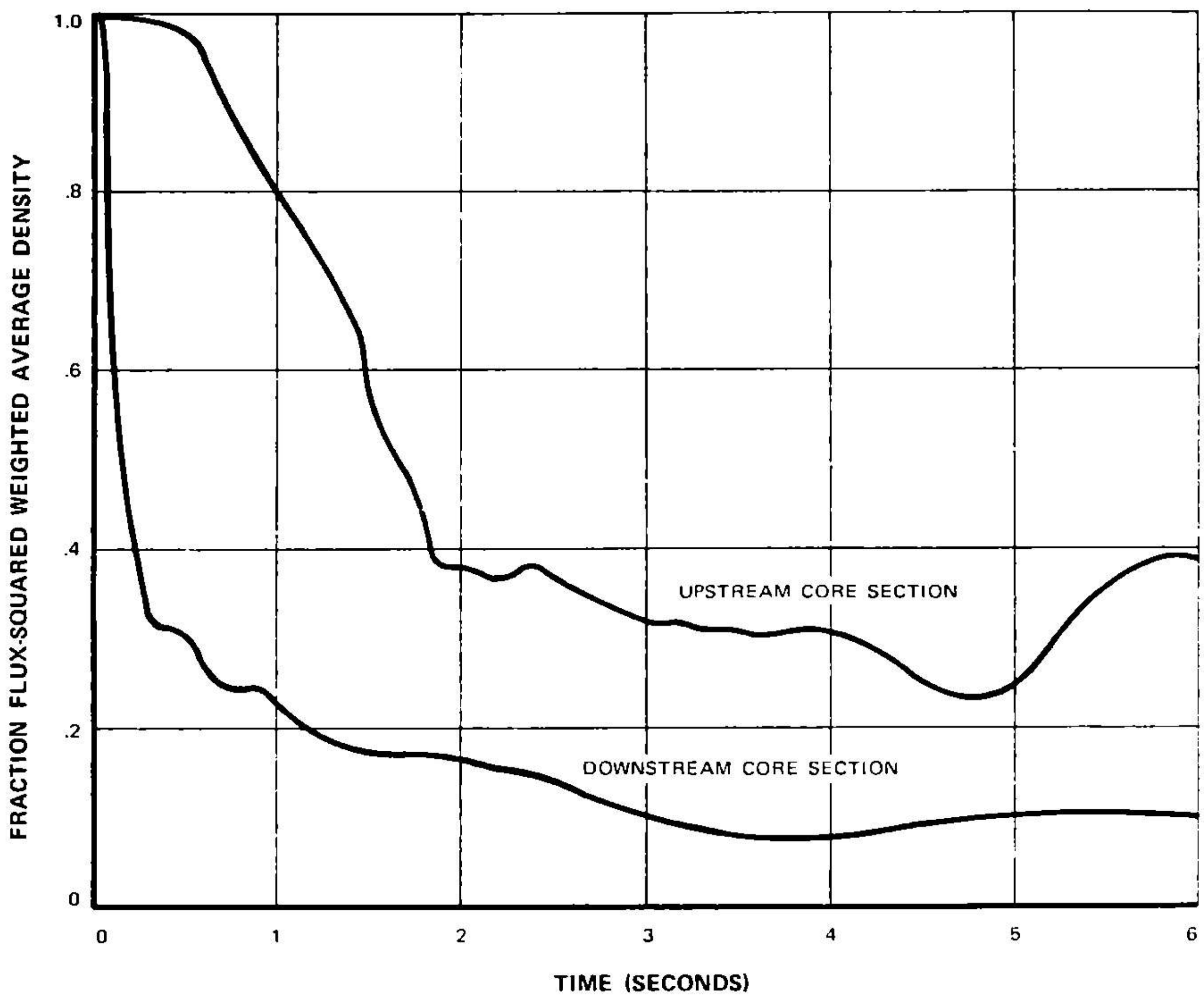

Fig. VI.3 Void Transients Following Maximum Inlet Header Break

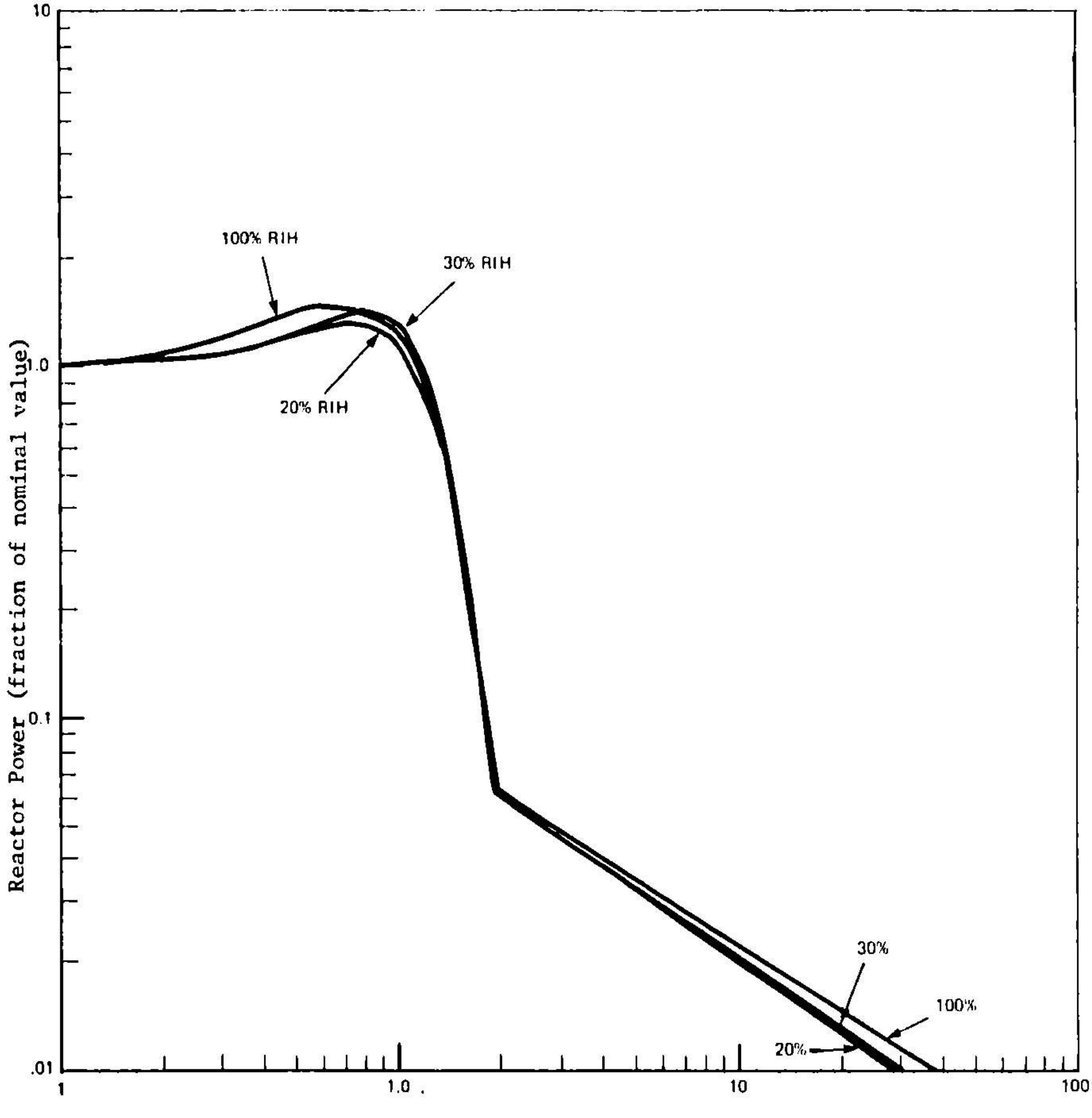

Fig. VI.4 Reactor Power versus Time Following an Inlet Header Break

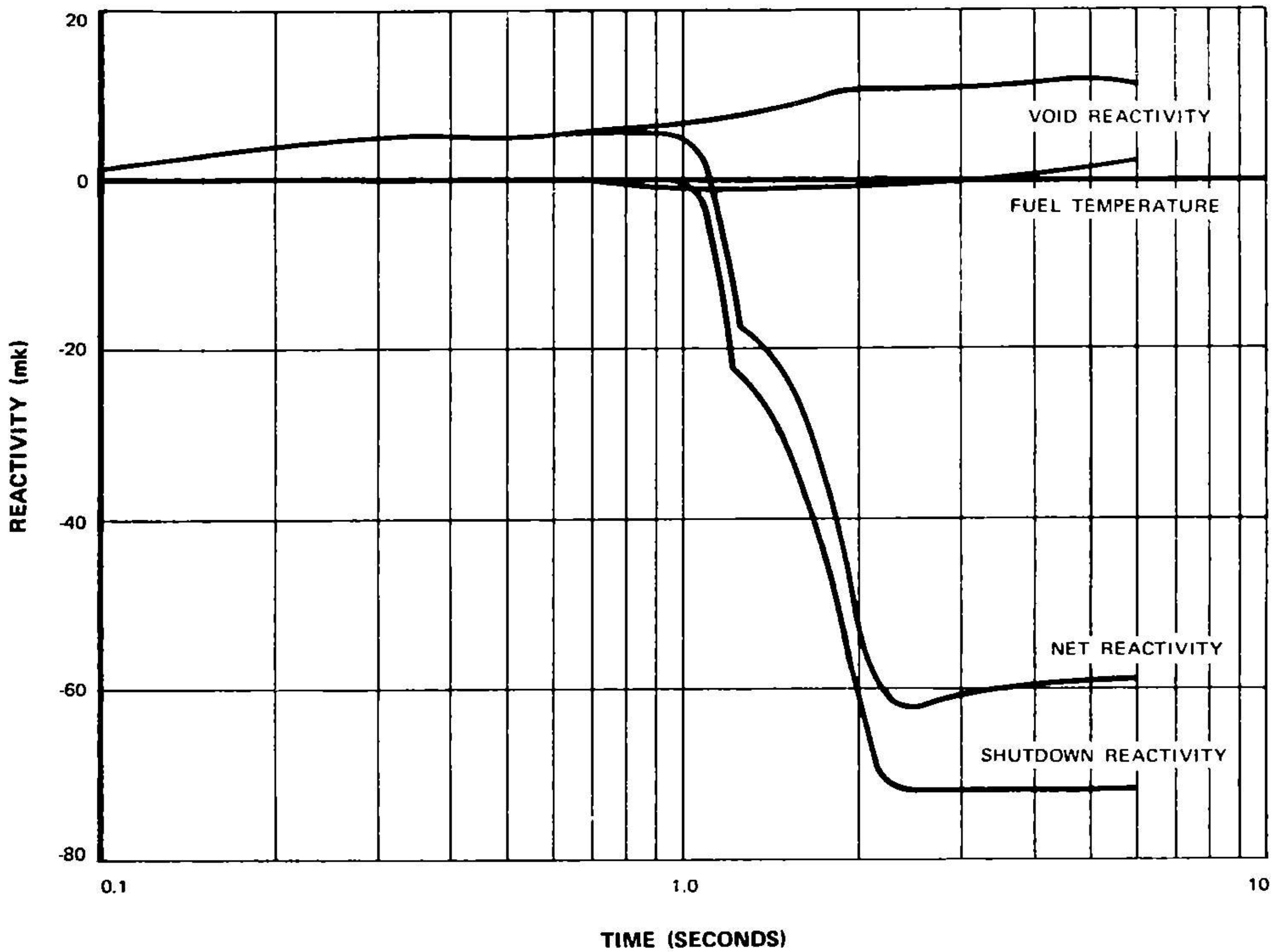

Fig. VI.5 Reactivity Transients Following Maximum Inlet Header Break (Fresh Fuel)

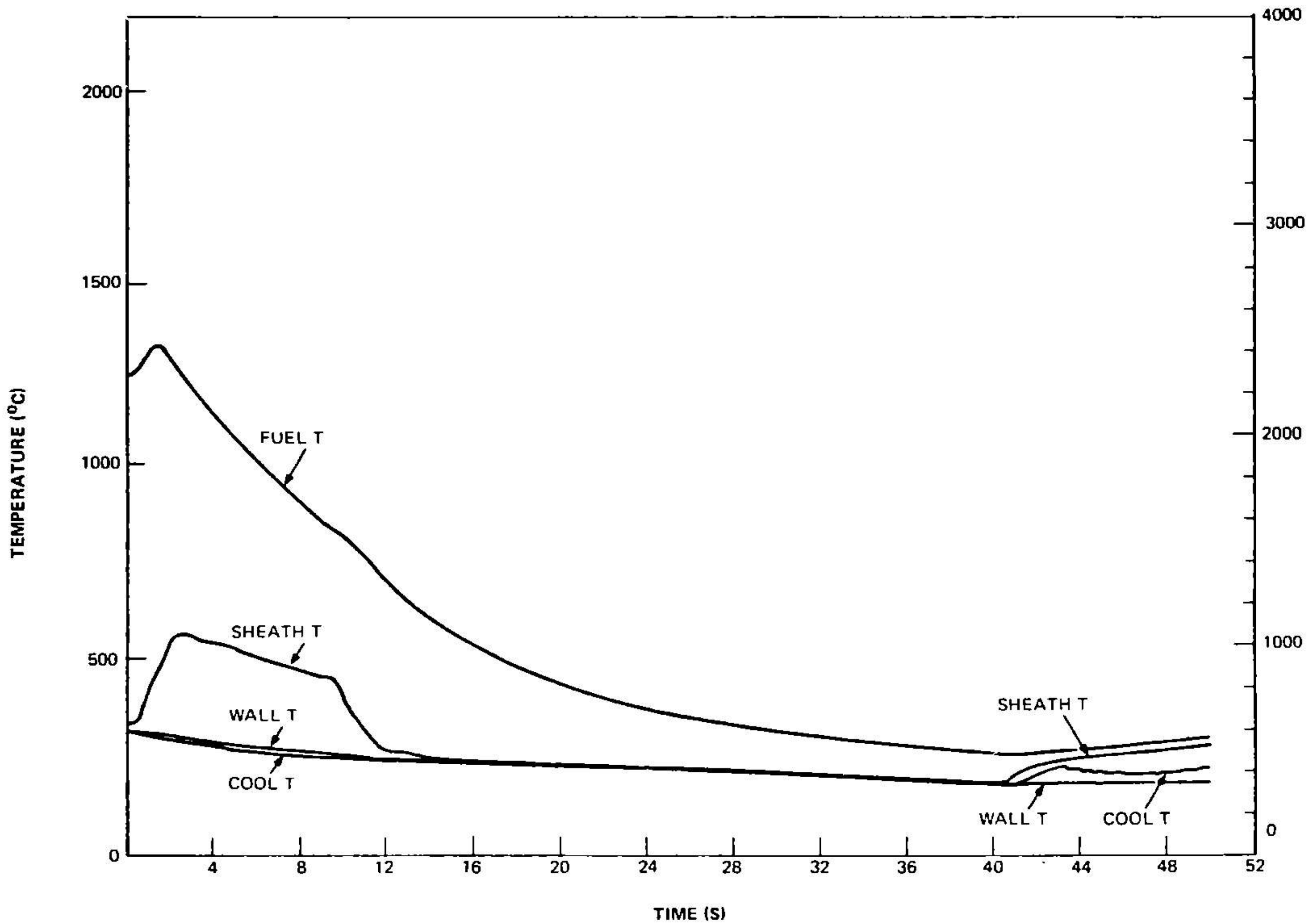

Fig. VI.6 Transient Temperatures in the Downstream Core Section for a 100% Inlet Header Break

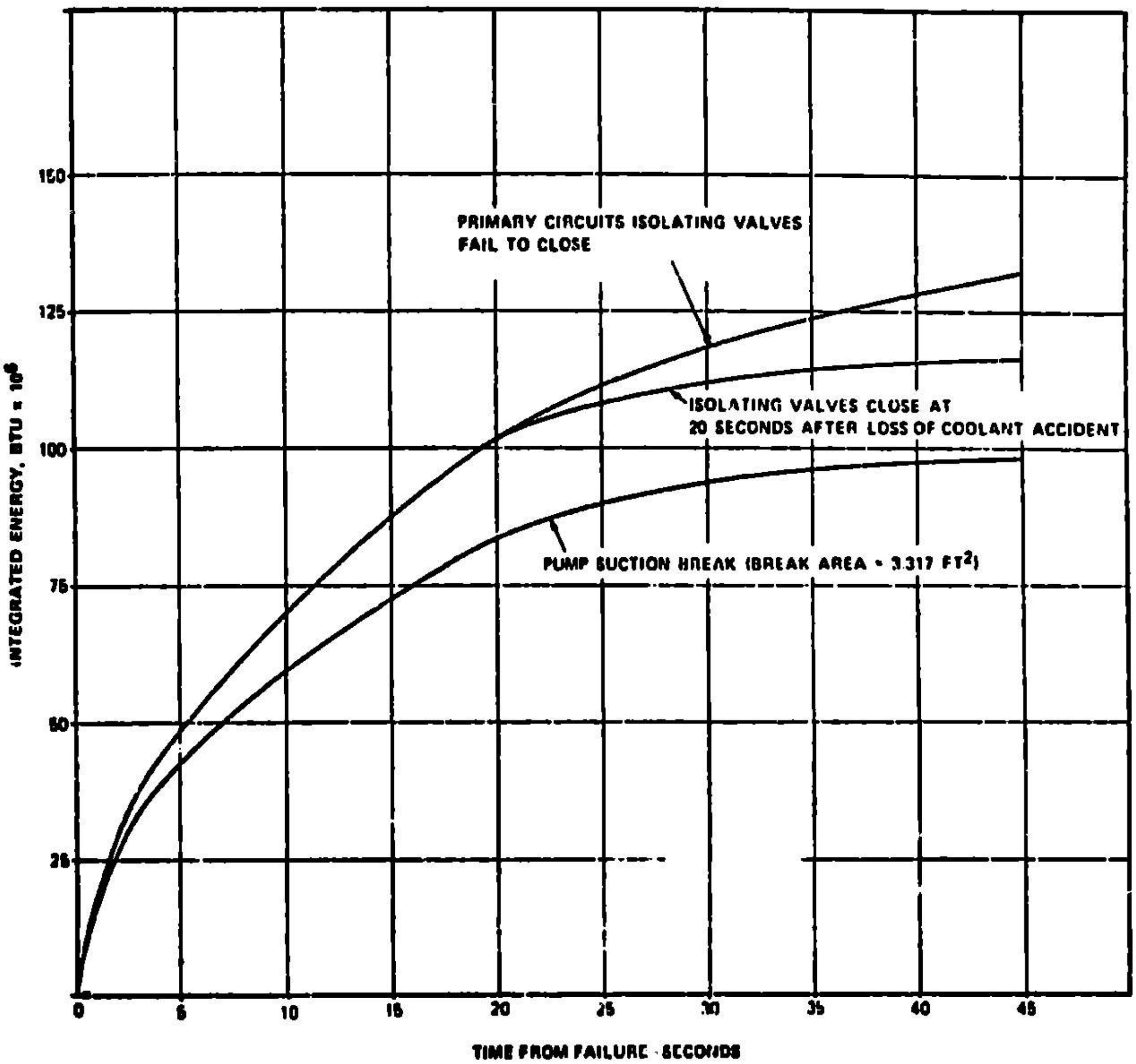

Fig. VI.7 Integrated Energy Versus Time Released into the Containment Following Maximum Pump Suction Header Break

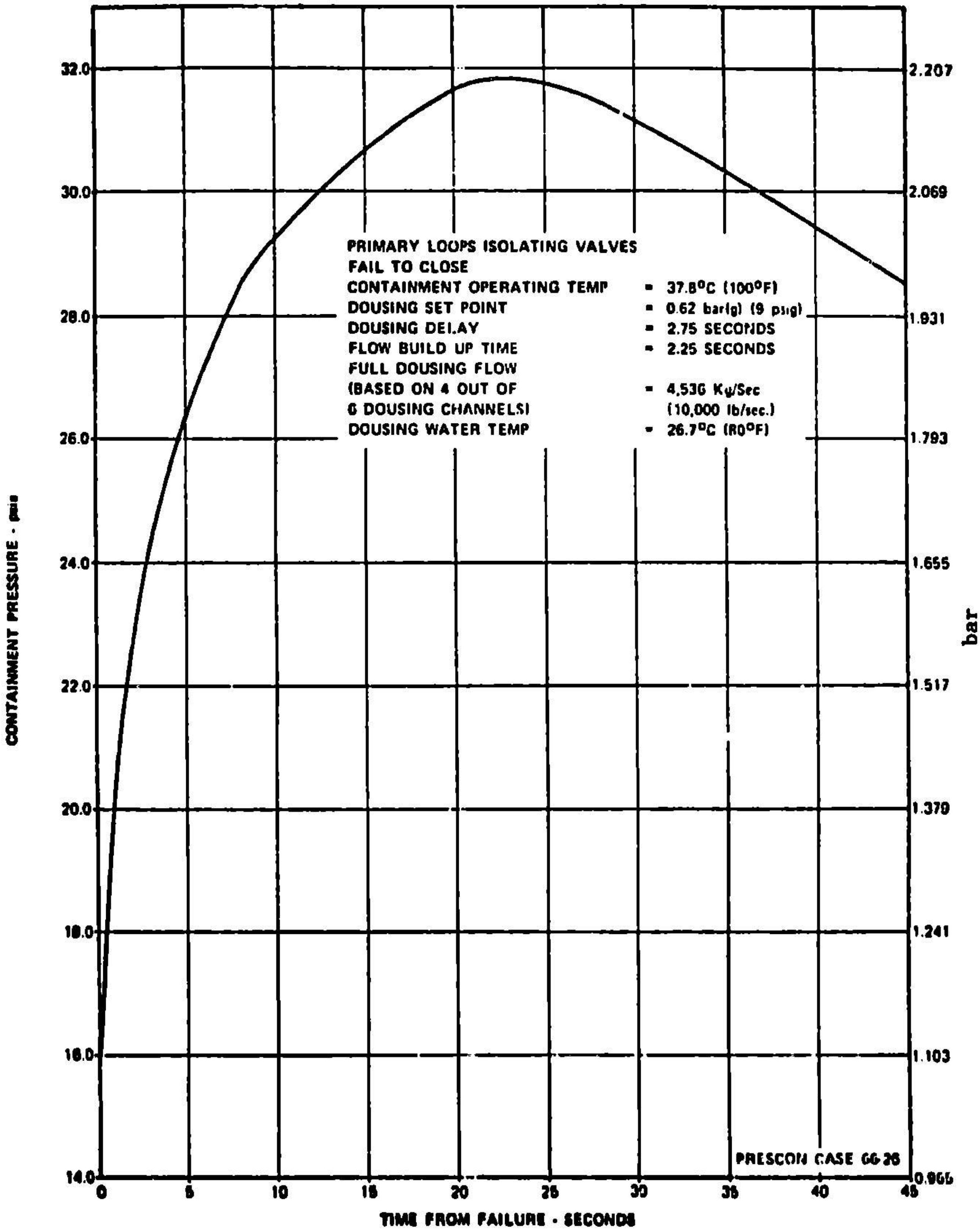

Fig. VI.8 Containment Pressure Transient Following Maximum Pump Suctions Header Break

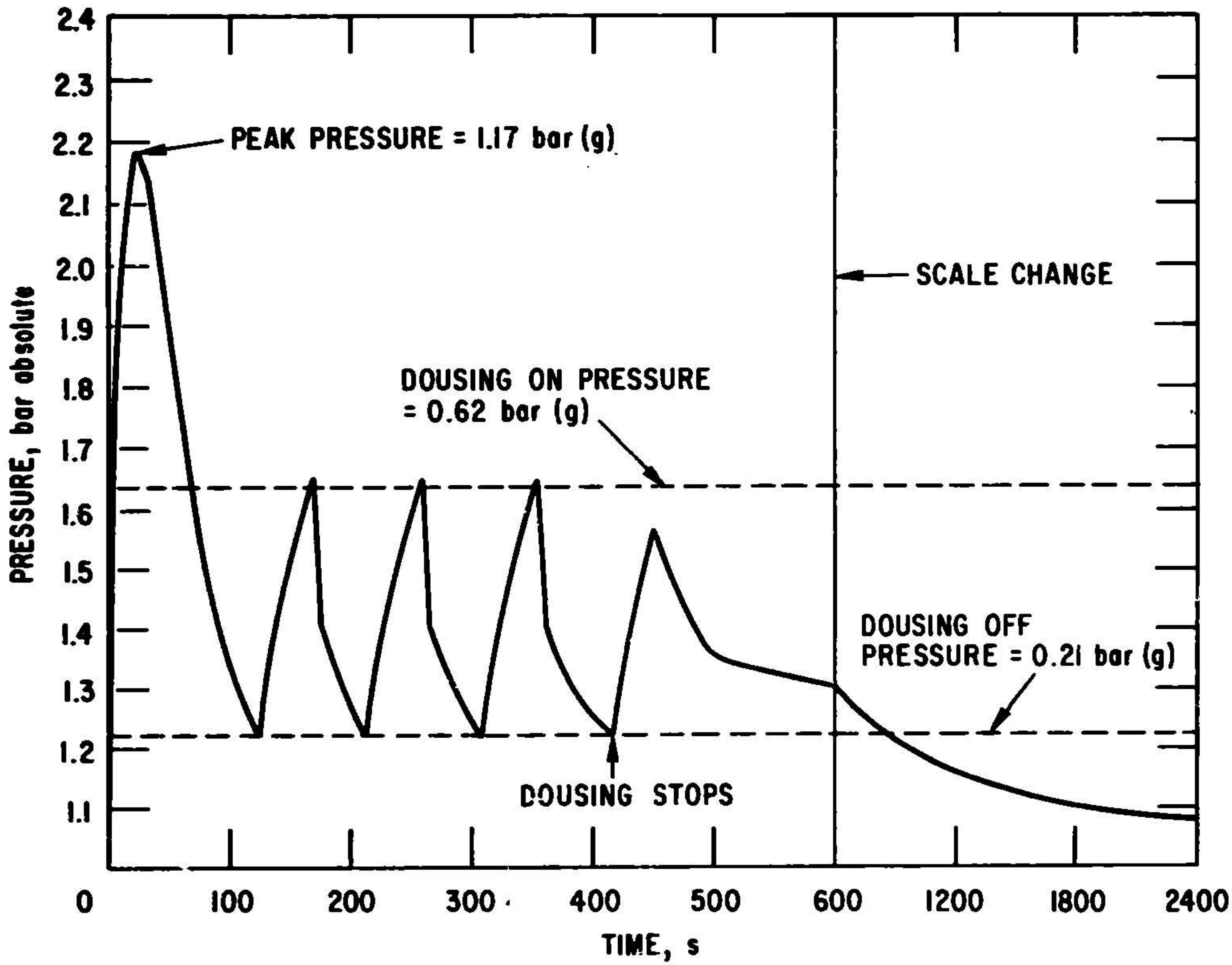

Fig. VI.9 Containment Pressure Versus Time for a Single-Unit Plant with Dousing System

VII. Evaluation of CANDU-PHW Power Plants with Respect to Current US-NRC General Design Criteria and Regulatory Guides

A. General Aspects

As was mentioned earlier, current US Regulatory Criteria and Guides have been, to a very large extent, evolved around the present generation of LWR nuclear power plants. It is therefore to be expected that introduction of a new type of nuclear power reactor requires careful examination of these criteria and guides in order to determine whether they are applicable to the new system, and, if so, to what degree, as well as to identify the need for possible changes. The CANDU-PHW reactor is not alone in this respect: Precedents do already exist for other nuclear reactor types having attained, or approaching, commercial application in the US, including the high temperature gas-cooled reactors (HTGRs), the liquid metal-cooled fast breeder reactors (LMFBRs), and the gas-cooled fast reactors (GCFRs).

Different reactor types have safety-related intrinsic characteristics which differ in varying degrees from each other. Some of the principal safety-related intrinsic characteristics of the CANDU-PHW reactor were discussed in Chapter III. It is clear that the differences in the safety-related intrinsic characteristics of the various reactor types should be reflected in the regulatory criteria, if the objective is to attain, at least to some degree of approximation, equivalency of safety, as discussed in Chapter I.

As was briefly discussed in Chapter I, there appears to be considerable merit in comparing the overall safety of different reactor systems on the basis of the results of probabilistic risk assessments. Even though, in the present state-of-the-art, such comparative risk assessments could probably not be carried out over the entire spectrum of accident initiators and sequences in the detail required, it seems that some tentative indications are already available for the CANDU-PHW reactor: One of the results of major importance, reported in the WASH-1400 study for LWRs, is the finding that any substantial release of radioactive material to the environment requires core meltdown as a primary (insufficient) condition. The presence in CANDU-PHW reactors of two large heat sinks, one of which is intimately dispersed throughout the core region (moderator), and one of which closely surrounds the entire core (light water shield), both provided with redundant cooling systems, would appear to make the probability of the occurrence of core melt accidents in CANDU-PHW reactors quite small. As was mentioned earlier in Chapter III, the ECIS does not, for CANDU-PHW reactors, constitute the final defense against core meltdown in case of a postulated large-scale LOCA; complete failure of the ECIS in a CANDU-PHW reactor, though resulting in core damage, does not lead to fuel melting, or to a loss of integrity of the power channel, since the decay heat can be removed through the moderator, having a heat capacity of 0.5 full-power-seconds/°C, and a redundant cooling system with heat removal capability equal to the decay heat. The light water shield, which has a heat capacity of 1.2 full-power-seconds/°C, and a heat removal capability of 0.3% of nominal power, constitutes in this respect a back-up for the moderator cooling system. Furthermore, reactor shutdown following a large-size LOCA, assuming no cooling at all, would result in boiling of the moderator and the light water shield after time periods of, respectively, $\sim$ 5 minutes and $\sim$ 40 minutes, during which the decay heat level would have decreased to, respectively, 3% and 1.8%.

In order to facilitate the evaluation, and to obtain a good overview, the US-NRC General Design Criteria with supporting Regulatory Guides will be subdivided, for the purposes of this study, in six categories (which may in some areas be overlapping to a certain extent) on the basis of the degree of applicability of the criteria to the CANDU-PHW reactor, and/or of the degree of conformance by the current CANDU-PHW reactor design to the criteria (see Table VII.1). It is noted that only Category VI comprises criteria for which modifications in the current CANDU-PHW reactor design (or applicable requirements) are recommended in case of a US-sited plant.

B. US-NRC General Design Criteria and Regulatory Guides

In this Section the US-NRC General Design Criteria (GDC) as given in Title 10 of the Code of Federal Regulations Part 50 (10-CFR-50), Appendix A, and as implemented by US-NRC Regulatory Guides, are evaluated with respect to their applicability to the CANDU-PHW reactor. Vice-versa the CANDU-PHW reactor is evaluated as regards its conformance to these criteria. As to the specific Regulatory Guides, and other supporting documents, used in the implementation of a specific General Design Criterion, reference is made to a companion study by United Engineers & Constructors on the same subject.[36]

Table VII.2 presents an overview of the primary results of this evaluation, in which the US-NRC General Design Criteria are subdivided in six categories as defined in Table VII.1.

GDC 1: Quality Standards and Records

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Evaluation of GDC 1

Canada has developed, and is applying, its own standards for nuclear power plants,[22-31] including those for Quality Assurance. The good performance of the CANDU-PHW reactors in Canada is testimony to this fact. There exist some differences between US and Canadian requirements and practices in this area. It appears, however, that in most aspects, if not all, the two Quality Assurance Programs are equivalent as regards their effect on safety. For a US-sited CANDU-PHW power plant, the US QA Program will have to be applied.

GDC 2: Design Bases for Protection Against Natural Phenomena

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for those structures, systems, and components shall reflect:

1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area with sufficient margin and for the limited accuracy, quantity, and period of time in which the historical data have been accumulated,

2) appropriate combinations of the effect of normal and accident conditions with the effects of the natural phenomena, and

3) the importance of the safety functions to be performed.

Evaluation of GDC 2

As discussed in Chapter V, CANDU-PHW reactors are designed to withstand the effects of natural phenomena, on the basis of the characteristics of the particular site in question. The requirements for seismic design in the US and Canada, though differing somewhat as regards certain aspects, appear to be equivalent relative to their effect on overall safety. A US-sited CANDU-PHW power plant would have to be designed and built in accordance with US requirements for protection against natural phenomena.

In order to maintain consistency within the US licensing environment, in particular with respect to nuclear reactor types already being licensed in the US, the level of seismic qualification of a number of safety-related systems may have to be changed for a US-sited CANDU-PHW power plant (see list of proposed design modifications given in Chapter VIII). As a trade-off, a number of safety-related seismically-qualified systems, provided in CANDU-PHW plants built in Canada, could be eliminated for a US-sited CANDU.

GDC 3: Fire Protection

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire-fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Evaluation of GDC 3

CANDU-PHW reactors are designed in accordance with strict fire protection requirements. Particular emphasis is placed in this respect on the design of the safety systems, since it is recognized that fires constitute a possible route to common-mode failures. Adequate separation is therefore provided, which has led to the Canadian practice of subdividing the various systems (process and safety-related) in two separate groups (see Table V.5), as well as the installation of a second control room at a separate location, with capability for reactor shutdown to cold conditions and availability of state-of-the-plant information.

The upgrading of the level of fire protection with respect to the use of fire-resistent materials for seals and for electric isolation of cables in the containment is consistent with the effort in the US in this area.

It is concluded that the CANDU-PHW reactor meets the intent of the requirements of GDC 3.

GDC 4: Environmental and Missile Design Bases

Structures, systems, and components important to safety shall be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant-accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

Evaluation of GDC 4

CANDU-PHW reactors meet the intent of the requirements of GDC 4, also as regards the effects of missiles, pipe whipping and discharging fluids.

It is observed that CANDU-PHW reactors are equipped with an extensive and sensitive leak detection system (also for economic reasons), which, in combination with the leak-before-break characteristics of the PHTS, renders the occurrence of a large-size LOCA extremely improbable. Notwithstanding this low probability, CANDU-PHW reactors are designed in accordance with strict requirements relative to the effects of postulated LOCAs. Large-diameter pipes are heavily anchored so as to minimize the possibility of damage propagation due to pipe break in case of a large-size break. Small-diameter pipes in CANDU-PHW reactors (specifically the feeders) are, however, not restrained to the same level required for LWRs, because, (1) analysis shows that the rupture of a feeder will not propagate to other feeders, (2) the feeders do not have much space to whip around, since they are closely packed together, and (3) too much anchoring of the feeders would inhibit thermal expansion and in-service inspection. It should be noted that US requirements relative to the restraining of piping have been specifically developed for the LWRs, where most PHTS piping is of the large diameter type and generally of greater length than in the CANDU-PHW reactor (PHTS piping in the CANDU-PHW reactor is kept as short as possible in order to minimize the inventory of heavy water). It would appear to be necessary to modify current US requirements in this area so as to be able to accommodate the intrinsic characteristics of the CANDU-PHW reactor which seem to preclude the possibility of restraining the feeders to the same degree as required for the large-diameter pipes in LWRs.

GDC 5: Sharing of Structures, Systems, and Components

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Evaluation of GDC 5

CANDU-PHW reactors are in conformance with this criterion.

GDC 10: Reactor Design

The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Evaluation of GDC 10

CANDU-PHW reactors are in conformance with this criterion.

GDC 11: <u>Reactor Inherent Protection</u>

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

<u>Evaluation of GDC 11</u>

The overall power-reactivity coefficient of CANDU-PHW reactors in the region of nominal operation is close to zero, and may (within the accuracy of calculation) be either slightly negative or slightly positive: For an increase in the reactor power the positive coolant-density-reactivity effect (including the effect due to an increase in the volume occupied by steam voids) is about equal to the negative fuel-temperature-reactivity effect.

Strictly speaking, the CANDU-PHW reactor does not, in view of the above, fully meet GDC 11. However, notwithstanding this fact, it would appear that there does <u>not</u> exist any problem deriving from GDC 11, that should prevent the licensing of CANDU-PHW reactors in the US. This opinion is supported by the following considerations:

1) The total amount of positive reactivity involved, even upon complete voiding of the PHTS of a CANDU-PHW reactor, is relatively small (∿ $1.5).

2) The use of pressure tubes in the core region of CANDU-PHW reactors permits subdivision of the PHTS into two or more separate sub-systems. Subdivision of the PHTS into two sub-systems reduces the total reactivity involved in the voiding of one sub-system to less than $1.00.

3) The positive reactivity that potentially could be introduced in a CANDU-PHW reactor by removal of the coolant from the core region constitutes the only pressure-dependent reactivity effect in CANDU-HWR reactors (compare with control rod ejection in LWRs, or steam void collapse in the core of BWRs); it is under nominal operating conditions not readily available.

4) The mean neutron lifetime of CANDU-PHW reactors is a factor 30 larger than that for LWRs, so that power excursions involving the same amount of reactivity are less severe in the former than in the latter type of reactor, particularly if close to, or in, the prompt critical range.

5) On the basis of equivalency of safety with other nuclear power reactor types, currently being licensed in the US (e.g., the LWRs), the CANDU-PHW reactor should be acceptable as regards GDC 11, since CANDU's level of conformance with the present wording of GDC 11 is consistent with that of other reactor types. The following points are relevant in this respect:

a) The volume of the steam voids in the core region of a BWR represents a total reactivity of ~ \$7;

b) In case of a postulated pressure transient in a BWR, due to, e.g., turbine trip or inadvertent actuation of the containment isolation valves, the prompt inherent feedback characteristics do not initially compensate for the increase in reactivity due to steam void collapse;

c) In the requirement of GDC 11 for a prompt inherent nuclear feedback, which "tends to compensate for a rapid increase in reactivity" is the implicit, but not stated, recognition that in LWRs the hydraulic forces existing in the core region during the blowdown phase of a LOCA may temporarily hold the control rods back from entering the core. Positive void-reactivity coefficients would therefore be not acceptable in LWRs. In CANDU-PHW reactors the neutron poison devices are installed in the low pressure moderator region, and are therefore not subject to the hydraulic forces du to LOCAs with breaks in the PHTS outside the core region. For LOCAs in CANDU-PHW reactors due to breaks in the core region (pressure tube rupture) only safety rods in the immediate vicinity of the break are affected; furthermore, the Second Shutdown System (SDS2), based on gadolinium injection into the moderator region, will not be affected at all. It should also be noted that CANDU-PHW reactors do not have ejectable neutron poison rods.

It would seem that, in view of the above considerations, it might be desirable to modify the present wording of GDC 11, so as to be less specifically aimed at the prompt inherent feedback for a particular type of reactivity insertion, and to be more generally aimed at the overall safety implications of inherent characteristics.

GDC 12: Suppression of Reactor Power Oscillations

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

Evaluation of GDC 12

The CANDU-PHW reactor is provided with the following systems:

1) three in-core neutron flux detection systems (1 for control, 2 for protection),

2) one zonal power control system, most parts of which are redundant,

3) two regional overpower protection systems, which are fully redundant and which include two different and diverse reactor shutdown systems, each having approximately 10 reactor trips for regional core protection.

Large CANDU-PHW cores may be subject to xenon-induced regional power oscillations if left uncontrolled. The CANDU-PHW reactor is, however, completely free from any hydraulic-induced instability (parallel-channel instability, etc.). The automatic zonal power control system is capable of maintaining the power everywhere in the core within safe limits.

On the basis of the above information it is concluded that the CANDU-PHW reactor fully meets the requirements of GDC 12.

Criterion 13: Instrumentation and Control

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

Evaluation of GDC 13

CANDU-PHW nuclear power plants are provided with extensive instrument and control systems, capable of monitoring those variables and systems that can affect the fission process, the integrity of the reactor core, the PHTS pressure boundary and the containment. Most control functions, including startup, are performed by two redundant computers. Protection functions are, however, not performed by the computers, there being strict separation between control and protection systems. The plant is provided with two separate control rooms in different locations, each with capability of shutting down and cooling the reactor to cold conditions, and providing continuous state-of-the-plant information to the operating staff (see also the evaluation under GDC 20 through GDC 25); this capability is still maintained in each control room even if total failure of all equipment in the other control room is assumed. It is concluded that CANDU-PHW nuclear power plants fully meet the requirements of GDC 13.

GDC 14: Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Evaluation of GDC 14

CANDU-PHW reactors are designed and constructed to the same quality level as the LWRs, through implementation of the ASME Boiler and Pressure Vessel Code, as supplemented by Canadian Standards[16-25] in areas not covered by ASME, as well as through implementation of the Canadian Quality Assurance Program. All high pressure components of the PHTS have "leak-before-break" characteristics. The plant is provided with extensive and sensitive leak detection systems (also for economic reasons). In view of this, the probability for a rapidly propagating failure in any part of the PHTS is extremely low. It is concluded that CANDU-PHW reactors meet the requirements of GDC 14 (see also the evaluation under GDC 30, 31, and 32).

GDC 15: Reactor Coolant System Design

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margins to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Evaluation of GDC 15

The reactor coolant pressure boundary of CANDU-PHW reactors is designed in accordance with ASME-Section III-Class 1 requirements, as supplemented by Canadian Standards[22-31] in areas not covered by ASME. It is concluded that CANDU-PHW reactors meet the requirements of GDC 15.

GDC 16: Containment Design

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

Evaluation of GDC 16

Canadian containment design practices for CANDU-PHW reactors differ from those in the US for LWRs in a number of areas. Though current containment systems for CANDU-PHW reactors no doubt meet the intent of GDC 16, it would appear that a number of design changes might be made for a US-sited CANDU power plant, in order to follow current US industrial practices. One of such design changes involves the use, in concrete containment systems, of a steel liner, in lieu of an epoxy liner which has been applied in some CANDU containments. It should be emphasized that in no way is it here suggested that an epoxy liner is not adequate; it is solely observed that it is current US practice to use steel liners in connection with concrete containment systems.

GDC 17: Electric Power Systems

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that 1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and 2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provision shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

Evaluation of GDC 17

CANDU-PHW reactors are provided with redundant and independent onsite and offsite power supplies, which meet the single failure criterion, and which are fully testable. Furthermore, two physically separate and redundant diesel-operated emergency power supplies are provided, one of which is seismically qualified to the level of the DBE.

It is concluded that CANDU-PHW reactors fully meet the intent of the requirements of GDC 17. However, US industrial and licensing practices appear to be somewhat different in a number of areas: As an example may be named that in the US it is customary to require only a single redundant diesel-operated emergency power supply.

GDC 18: Inspection and Testing of Electrical Power Systems

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically 1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and 2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Evaluation of GDC 18

CANDU-PHW reactors fully meet the requirements for inspection and testability of GDC 18.

GDC 19: Control Room

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant-accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided 1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and 2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Evaluation of GDC 19

CANDU-PHW reactors are provided with a main control room and an auxiliary control room; the latter is installed in a separate location. Both control rooms have full capability for reactor shutdown to cold conditions and for providing continuous state-of-the-plant information; this capability is still maintained in each control room, even if total failure of all equipment in the other control room is assumed.

CANDU-PHW reactors fully meet the intent of the requirements of GDC 19. However, again in certain areas there are differences between Canadian and US industrial and licensing practices: An example is the US requirement for a control room ventilation system having a remote air intake to enhance the chances for maintaining continuous habitability of the main control room under accident conditions. The Canadian position in this respect is that the auxiliary control room in the separate location would be used for the low probability event that the main control room would suffer an impaired habitability. Another example concerns different practices in Canada and the US relative to personnel security provisions for the main control room. None of these differences are of major importance, and could therefore be easily accommodated for a US-sited CANDU-PHW power plant, if necessary.

GDC 20: Protection System Functions

The protection system shall be designed 1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and 2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Evaluation of GDC 20

CANDU-PHW reactors are provided with a protection system that fully meets the requirements of GDC 20.

GDC 21: Protection System Reliability and Testability

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that 1) no single failure results in loss of the protection function and 2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of the functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Evaluation of GDC 21

The protection system of CANDU-PHW reactors contains two separate shutdown systems (SDS1 and SDS2), based on diverse types of neutron poison devices. Each shutdown system is required to have an unreliability (unavailability) of less than 10^{-3}, and should be testable during reactor operation.

CANDU-PHW reactors fully meet the requirements of GDC 21.

GDC 22: Protection System Independence

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Evaluation of GDC 22

The protection systems of CANDU-PHW reactors are designed to assure that possible effects of normal operation, maintenance, testing and postulated accident conditions do not impair the protection functions. There is complete independence between redundant portions of the various subsystems. Each channel is entirely self-contained and has its own power supply. Failure of a power supply results in a trip condition for the affected channel.

It is concluded that the CANDU-PHW reactor fully meets the requirements of GDC 22.

GDC 23: Protection System Failure Modes

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as discontinuation of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Evaluation of GDC 23

The protection system of CANDU-PHW reactors fully meets the requirements of GDC 23. The upgrading of the level of fire protection with respect to the use of fire-resistant materials for electric isolation of cables is consistent with the effort in the US in this area.

GDC 24: Separation of Protection and Control Systems

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Evaluation of GDC 24

There is complete separation between control and protection functions in CANDU-PHW reactors. This separation extends to the neutron poison devices, in that separate and diverse neutron poison devices are provided for control and protection functions; thus the level of separation between control and protection systems exceeds that in current LWRs. Most control functions are performed by two redundant computers. The protection system consists, however, entirely of hard-wired circuitry and is fully self-contained, also as regards monitoring and alarming functions. For operational convenience, some of the safety-related data is also printed out by the computers; data channels from the protection system to the computers are fully buffered; there is no feedback from the computers into the protection system.

It is concluded that CANDU-PHW reactors fully meet the requirements of GDC 24.

GDC 25: Protection System Requirements for Reactivity Control Malfunctions

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Evaluation of GDC 25

CANDU-PHW reactors fully meet the requirements of GDC 25.

GDC 26: Reactivity Control System Redundancy and Capability

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

Evaluation of GDC 26

CANDU-PHW reactors are provided with four systems for neutron poison control, namely: (1) control rods, (2) adjuster rods, (3) light water cells, and (4) neutron poison (boron) addition into the moderator region. In addition there are two reactor shutdown systems, namely (1) shutdown rods (SDS1), and (2) gadolinium injection into the moderator region (SDS2). Together these systems provide ample redundancy and capability for all normal and postulated off-normal plant conditions.

It is concluded that CANDU-PHW reactors fully meet the requirements of GDC 26.

GDC 27: Combined Reactivity Control Systems Capability

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Evaluation of GDC 27

CANDU-PHW reactors fully meet the requirements of GDC 27.

GDC 28: Reactivity Limits

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure and cold water addition.

Evaluation of GDC 28

CANDU-PHW reactors fully meet the requirements of GDC 28 (see also Chapter VI, point B: LORAs).

All neutron poison devices in CANDU-PHW reactors are installed in the low pressure moderator region; this obviates consideration of control rod ejection accidents. Control rods are inserted from above into the core by gravity; rod dropout accidents do not, therefore, apply to CANDU-PHW reactors.

GDC 29: Protection Against Anticipated Operational Occurrences

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Evaluation of GDC 29

CANDU-PHW reactors fully meet the requirements of GDC 29.

GDC 30: Quality of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

Evaluation of GDC 30

The entire PHTS is designed and fabricated in accordance with ASME-Section III-Class 1 requirements as complemented by Canadian Nuclear Standards in those areas not at present covered by ASME.

CANDU-PHW reactors are provided with extensive leak detection systems, including (1) moisture and vapor recovery systems, detecting the presence of leaks, both locally and overall, (2) a gas annulus surveillance system, detecting leaks in pressure tubes, (3) an ultrasonic sound pick-up system installed on the head of the fueling machine, detecting the presence of cracks in pressure tubes and/or endfittings, etc. It should be noted that economic considerations preclude operation with leaks of any magnitude in a D_2O-cooled system.

It is concluded that CANDU-PHW reactors meet the requirements of GDC 30.

GDC 31: Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state, and transient stresses, and (4) size of flaws.

Evaluation of GDC 31

As mentioned under GDC 14, CANDU-PHW reactors are designed and constructed to the same quality level as the LWRs, through implementations of the ASME B&PV Code, as supplemented by Canadian Standards[22-31], as well as through implementation of the Canadian Quality Assurance Program. There are three noteworthy departures from the ASME B&PV Code, namely (1) Zr-Nb alloy is used for the pressure tubes, (2) special type 403 SS is used for the end-fittings of the pressure tubes, and (3) pressure tubes and endfittings are connected together by rolled joints.

The use of non-ferrous materials as part of the primary coolant pressure boundary of nuclear power plants is not covered by the ASME B&PV Code, primarily because there exists no need for this in LWRs. Non-ferrous materials are, however, used as part of the PHTS on a world-wide basis in all countries operating pressure tube reactors, including the USSR, UK, France, W. Germany, Italy, Japan, India, etc. There appears to exist no inherent reason why Zr-Nb alloy should not be acceptable for use in the PHTS.

ASME Code Case 1337-10 (April 28, 1975), entitled "Special Type 403 Modified Forgings and Bars" addresses the use of special type 403 SS.

The use of rolled joints as part of the PHTS of nuclear power plants is similarly not covered by the ASME B&PV Code, since no need exists for this in LWRs. Extensive tests and operating experience in Canada and elsewhere (USSR, France, UK, West Germany, Italy, Japan, India, etc.) have shown that this kind of joint can be fabricated to high standards so as to constitute a reliable part of the PHTS.

None of the above departures from the ASME B&PV Code appear to be of a nature that would preclude their later incorporation in the code, provided sufficient interest were to exist in the US.

GDC 32: Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit: (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity; and (2) an appropriate material surveillance program for the reactor pressure vessel.

Evaluation of GDC 32

Canada has developed an extensive Inservice Inspection Program[23-31] covering all vital parts of the reactor coolant pressure boundary. In addition to the standard techniques, a number of special techniques have been developed, including:

(1) ultrasonic inspection of the pressure tubes and rolled joints can be performed during operation by means of the fueling machine;

(2) ultrasonic and eddy current full volumetric inspection (including wall-thickness measurements) of the pressure tubes and rolled joints can be performed during shutdown;

(3) surface condition inspection of pressure tubes (wear, scratches, corrosion, etc.) can be performed during shutdown using profilometry, photography, etc.

(4) bow, diameterical changes, length increase, hardness, etc. of pressure tubes can be measured during shutdown, etc.

A material surveillance program using coupons for the pressure tubes and endfittings is not practical since the material not only has to be exposed to the neutron flux (which could be done by placing coupons in the moderator region), but also to the coolant at high temperatures so as to be subject also to hydrogen embrittlement. It is suggested instead that, if needed, at appropriate points during the reactor's life a pressure tube be taken out and sacrificed for material testing surveillance purposes.

GDC 33: Reactor Coolant Makeup

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

Evaluation of GDC 33

CANDU-PHW reactors meet the intent of the requirements of GDC 33. For a US-sited plant some changes may be required in order to follow US industrial and licensing practices.

GDC 34: Residual Heat Removal

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Evaluation of GDC 34

The Shutdown Cooling System of CANDU-PHW reactors consists of two 100% capacity pumps and two 50% capacity heat exchangers to remove decay heat following reactor shutdown; it is designed for the full nominal operating pressure of the PHTS ($\sim$1450 psi), so that it can, if needed, be connected to the PHTS immediately following reactor shutdown without first requiring a PHTS cooldown. The pumps are supplied from different power sources. Due to the different Canadian position with regard to required redundancy for passive low-pressure components, some of the piping on the secondary side of the Shutdown Cooling System is not at the redundancy level required in the U.S.

Decay heat can also be removed through the steam generators, by means of natural circulation on the primary side and by feeding the steam generators on the secondary side either from the auxiliary feedwater supply or from the EWS.

It is concluded that CANDU-PHW reactors meet the intent of the requirements of GDC 34, surpassing even in some respects (e.g., design of the Shutdown Cooling System for full nominal PHTS pressure) the capabilities of the counterpart systems in LWRs. Due to differences in design approach in Canada and the U.S., some changes may be required for a U.S.-sited CANDU-PHW power plant.

GDC 35: Emergency Core Cooling

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Evaluation of GDC 35

CANDU-PHW reactors meet the intent of the requirements of GDC 35. There are, however, some differences between the design requirements for the ECIS of CANDU-PHW reactors in Canada and those for the ECCS of LWRs in the U.S. with respect to the maximum allowable cladding temperature reached during a LOCA (see Chapter V, Section C), as well as with respect to the required redundancy of low-pressure passive components.

It would appear that for a U.S.-sited CANDU-PHW power plant, some design changes may have to be made in order to follow U.S. industrial and licensing practices.

It is noted here that the ECIS in CANDU-PHW reactors does not constitute the final defense against core meltdown (as is the case in LWRs): The moderator cooling system is a diverse back-up for the ECIS, capable of preventing fuel melting, and maintaining a coolable core configuration.

GDC 36: Inspection of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping to assure the integrity and capability of the system.

Evaluation of GDC 36

CANDU-PHW reactors fully meet the requirements of the GDC 36.

GDC 37: Testing of Emergency Core Cooling System

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leak-tight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal, and emergency power sources, and the operation of the associated cooling water system.

Evaluation of GDC 37

CANDU-PHW reactors fully meet the requirements of GDC 37.

GDC 38: Containment Heat Removal

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant-accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable inter-connections, lead detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Evaluation of GDC 38

The Containment Heat Removal function in CANDU-PHW reactors is performed by four separate systems, namely (1) the dousing system, (2) the building cooling system consisting of 10 to 15 separate units with air-to-water heat exchangers, (3) the ECC recovery system, and (4) the steam generators via feed from the EWS.

CANDU-PHW reactors meet fully the intent of GDC 38. However, for a U.S.-sited plant some changes may be required in order to follow U.S. industrial and licensing practices.

GDC 39: <u>Inspection of Containment Heat Removal System</u>

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components.

<u>Evaluation of GDC 39</u>

CANDU-PHW reactors fully meet the requirements of GDC 39.

GDC 40: Testing of Containment Heat Removal System

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure: (1) the structural and leaktight integrity of its components; (2) the operability and performance of the active components of the system; and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Evaluation of GDC 40

CANDU-PHW reactors fully meet the requirements of GDC 40.

GDC 41: Containment Atmosphere Cleanup

Systems to control fission products, hydrogen, oxygen, and other substances which may be relased into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), its safety function can be accomplished, assuming a single failure.

Evaluation of GDC 41

CANDU-PHW reactors meet the intent of the requirements of GDC 41. However, in view of differences in design and licensing approach in Canada and the US with respect to containment atmosphere cleanup some changes may be required for a US-sited plant.

GDC 42: Inspection of Containment Atmosphere Cleanup Systems

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

Evaluation of GDC 42

CANDU-PHW reactors fully meet the requirements of GDC 42.

GDC 43: <u>Testing of Containment Atmosphere Cleanup Systems</u>

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

<u>Evaluation of GDC 43</u>

CANDU-PHW reactors fully meet the requirements of GDC 43.

GDC 44: Cooling Water

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Evaluation of GDC 44

CANDU-PHW reactors are provided with two separate cooling water sources (see Figure V.1), namely the Service Water Supply, and the Emergency Water Supply (EWS), the latter being powered from the Emergency Power Supply (EPS). As indicated in Figure V.1, the EWS can be connected to 1) the ECCS, 2) the PHTS (after depressurization), and 3) the steam generators (boilers).

CANDU-PHW reactors fully meet the intent of the requirements of GDC 44. However, in view of the differences between Canadian and US industrial and licensing practices, it is expected that US-sited CANDU-PHW power plants would probably have to undergo some modifications in this area.

GDC 45: Inspection of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

Evaluation of GDC 45

CANDU-PHW reactors fully meet the requirements of GDC 45.

GDC 46: Testing of Cooling Water System

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure 1) the structural and leaktight integrity of its components, 2) the operability and the performance of the active components of the system, and 3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant-accidents including operation of applicable portions of the protection system, and the transfer between normal and emergency power sources.

Evaluation of GDC 46

CANDU-PHW reactors fully meet the requirements of GDC 46.

GDC 50: Containment Design Basis

The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of 1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, 2) the limited experience and experimental data available for defining accident phenomena and containment responses, and 3) the conservatism of the calculational model and input parameters.

Evaluation of GDC 50

CANDU-PHW reactors fully meet the intent of the requirements of GDC 50, in that 1) the effects of all potential energy sources have been considered in the design, and 2) considerable conservatism is being observed in all design aspects. However, in view of some differences between Canadian and US industrial and licensing practices, it is expected that, in case of a US-sited CANDU-power plant, a containment design approach would be followed, which would be in line with current US practices.

GDC 51: Fracture Prevention of Containment Pressure Boundary

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing and postulated accident conditions 1) its ferritic materials behave in a non-brittle manner; and 2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining: 1) material properties, 2) residual, steady-state, and transient stresses, and 3) size of flaws.

Evaluation of GDC 51

CANDU-PHW reactors have been built with essentially two types of containment, namely multi-unit and single-unit containment systems. Both types of containment use concrete as the main structural material. Some of these containment systems have been provided with epoxy liners, whereas others have been equipped with steel liners. Canadian experience has shown that both liners are capable of satisfactory performance. In Canada the choice of type of liner for the containment system is treated on the basis of customer's preference.

It is concluded that the CANDU-PHW reactor meets the intent of the requirements of GDC 51. However, for U.S.-sited CANDU-PHW reactors, it will probably be necessary to limit consideration only to containments having steel liners, in view of existing U.S. industrial and licensing practices for LWRs.

GDC 52: Capability for Containment Leakage Rate Testing

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

Evaluation of GDC 52

CANDU-PHW reactors fully meet the requirements of GDC 52.

GDC 53: Provisions for Containment Testing and Inspection

The reactor containment shall be designed to permit 1) appropriate periodic inspection of all important areas, such as penetration, 2) an appropriate surveillance program, and 3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

Evaluation of GDC 53

CANDU-PHW reactors fully meet the requirements of GDC 53.

GDC 54: Piping Systems Penetrating Containment

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Evaluation of GDC 54

CANDU-PHW reactors meet the intent of the requirements of GDC 54. Some modifications may be desirable for a US-sited plant, in view of differences between Canadian and US industrial and licensing practices, in particular as regards the number and the location of the required isolation valves. These changes, if any, are expected to be not cost-significant.

GDC 55: Reactor Coolant Pressure Boundary Penetrating Containment

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines are acceptable on some other defined basis:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accident rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as a higher quality in design, fabrication, and testing, additional provisions for in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

Evaluation of GDC 55

CANDU-PHW reactors of the standard 600 MWe type do not have any reactor coolant pressure boundary penetrating the containment. It is therefore concluded that GDC 55 does not apply to CANDU-PHW reactors.

GDC 56: Primary Containment Isolation

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Evaluation of GDC 56

CANDU-PHW reactors meet the intent of the requirements of GDC 56 since they are provided with dual valves for all lines penetrating the containment. In general, however, both valves are installed outside of the containment, which is different from the U.S. practice.

It would appear that some changes are required in this area for a U.S.-sited CANDU-PHW reactor to accommodate differences between US and Canadian practices. These changes are expected to be not cost-significant.

GDC 57: Closed System Isolation Valves

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

Evaluation of GDC 57

The main steam lines of CANDU-PHW reactors are not equipped with isolation valves, because there is no need for this in view of the fact that the steam is not radioactive. Furthermore, steam line isolation would prevent the rapid safety cooldown function (i.e., blowdown of the secondary side of the steam generators to the atmosphere), that is initiated immediately following a LOCA with the aim of obtaining a rapid depressurization of the PHTS so as to speed up entry of coolant from the ECCS for small-break LOCAs.

CANDU-PHW reactors meet the intent of GDC 57. Some modifications may be desirable for a US-sited plant, in view of differences between Canadian and US industrial and licensing practices, in particular as regards the number and the location of the required isolation valves. These modifications are expected to be not cost-significant.

GDC 60: Control of Releases of Radioactive Materials to the Environment

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

Evaluation of GDC 60

CANDU-PHW reactors meet fully the intent of the requirements of GDC 60.

GDC 61: Fuel Storage and Handling and Radioactivity Control

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed: 1) with a capability to permit appropriate periodic inspection and testing of components important to safety, 2) with suitable shielding for radiation protection, 3) with appropriate containment, confinement and filtering systems, 4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and 5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Evaluation of GDC 61

CANDU-PHW reactors have been built with various types of liners (epoxy, steel, aluminum) in the spent fuel pool on the basis of customer's preference. All types of liners have performed well. US practice is to use only steel for liners in spent fuel pools.

CANDU-PHW reactors meet the intent of the requirements of GDC 61. For a U.S.-sited CANDU-PHW plant some changes may be required to reflect differences in U.S. industrial and licensing practices.

GDC 62: Prevention of Criticality in Fuel Storage and Handling

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

Evaluation of GDC 62

CANDU-PHW reactors use natural uranium fuel. The fresh fuel is stored in a dry vault. The spent fuel is stored in a bay filled with light water. It is not possible to achieve criticality using natural uranium fuel in conjunction with light water as a moderator. Therefore criticality outside the reactor is not a problem for CANDU-PHW reactor fuel, so that GDC 62 is certainly met.

GDC 63: Monitoring Fuel and Waste Storage

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas 1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and 2) to initiate appropriate safety actions.

Evaluation of GDC 63

CANDU-PHW reactors meet the requirements of GDC 63.

GDC 64: Monitoring Radioactivity Releases

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant-accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Evaluation of GDC 64

CANDU-PHW reactors meet the requirements of GDC 64.

TABLE VII.1: Categorization of US-NRC General Design Criteria

Category	Degree of Applicability of the US-NRC Criteria to the Current CANDU-PHW Reactor Design, and/or Degree of Conformance of the Current CANDU-PHW Reactor Design to US-NRC Criteria
I	US-NRC criteria that are not applicable to the CANDU-PHW reactor, and which for that reason do not have to be met for a US-sited CANDU-PHW plant.
II	US-NRC criteria that are fully met by present-design CANDU-PHW reactors.
III	US-NRC criteria that are met relative to intent by CANDU-PHW reactors (but not with respect to the exact wording), and for which no change in the current CANDU-PHW design is required for a US-sited plant, to meet the requirement of equivalency of safety with LWRs, because the wording may be too specifically aimed at LWRs, or in view of the differences in safety-related intrinsic characteristics between CANDU-PHW reactors and LWRs.
IV	US-NRC criteria that are met relative to intent by CANDU-PHW reactors (but not with respect to the exact wording), and for which no change in the current CANDU-PHW design is required for a US-sited plant, but for which an ASME B&PV code case (modification and/or extension) may be required for a US-sited plant, in view of the CANDU-PHW reactor's specific design characteristics.

TABLE VII.1 (Contd.)

Category	Degree of Applicability of the US-NRC Criteria to the Current CANDU-PHW Reactor Design, and/or Degree of Conformance of the Current CANDU-PHW Reactor Design to US-NRC Criteria
V	US-NRC criteria that are <u>not met</u> by CANDU-PHW reactors, and for which <u>conformance is not necessary</u> to meet the requirement of equivalency of safety with LWRs, in view of the differences in safety-related intrinsic characteristics between CANDU-PHW reactors and LWRs.
VI	US-NRC criteria that are <u>met relative to intent</u> by CANDU-PHW reactors (but not with respect to the exact wording), but for which a <u>change</u> in the current CANDU-PHW design (or applicable requirements) <u>may be necessary</u> for a US-sited plant.

TABLE VII.2 Category of Applicability of US-NRC Criteria and/or Conformance by CANDU-PHW Reactor

US-NRC Criteria 10-CFR-50 Appendix A	Categories					
	I	II	III	IV	V	VI
1. Quality Standards and Records						X
2. Design Bases for Protection Against Natural Phenomena						X
3. Fire Protection		X				
4. Environmental and Missile Design Bases					X	
5. Sharing of Structures, Systems and Components		X				
10. Reactor Design		X				
11. Reactor Inherent Protection					X	
12. Suppression of Reactor Power Oscillations		X				
13. Instrumentation and Control		X				
14. Reactor Coolant Pressure Boundary			X			
15. Reactor Coolant System Design		X				
16. Containment Design						X
17. Electric Power Systems						X
18. Inspection and Testing of Electric Power Systems		X				
19. Control Room						X
20. Protection System Functions		X				

TABLE VII.2 (Contd.)

US-NRC Criteria 10-CFR-50 Appendix A	Categories					
	I	II	III	IV	V	VI
21. Protection System Reliability and Testability		X				
22. Protection System Independence		X				
23. Protection System Failure Modes		X				
24. Separation of Protection and Control Systems		X				
25. Protection System Requirements for Reactivity Control Malfunctions		X				
26. Reactivity Control System Redundancy and Capability		X				
27. Combined Reactivity Control Systems Capability		X				
28. Reactivity Limits		X				
29. Protection Against Anticipated Operational Occurrences		X				
30. Quality of Reactor Coolant Pressure Boundary		X				
31. Fracture Prevention of Reactor Coolant Pressure Boundary				X		
32. Inspection of Reactor Coolant Pressure Boundary				X		
33. Reactor Coolant Makeup						X
34. Residual Heat Removal						X

TABLE VII.2 (Contd.)

US-NRC Criteria 10-CFR-50 Appendix A	Categories I	II	III	IV	V	VI
35. Emergency Core Cooling						X
36. Inspection of Emergency Core Cooling System		X				
37. Testing of Emergency Core Cooling System		X				
38. Containment Heat Removal						X
39. Inspection of Containment Heat Removal System		X				
40. Testing of Containment Heat Removal System		X				
41. Containment Atmosphere Cleanup						X
42. Inspection of Containment Atmosphere Cleanup Systems		X				
43. Testing of Containment Atmosphere Cleanup Systems		X				
44. Cooling Water						X
45. Inspection of Cooling Water System		X				
46. Testing of Cooling Water System		X				
50. Containment Design Basis						X
51. Fracture Prevention of Containment Pressure Boundary						X
52. Capability for Containment Leakage Rate Testing		X				

TABLE VII.2 (Contd.)

US-NRC Criteria 10-CFR-50 Appendix A	Categories					
	I	II	III	IV	V	VI
53. Provisions for Containment Inspection and Testing		X				
54. Systems Penetrating Containment						X
55. Reactor Coolant Pressure Boundary Penetrating Containment	X					
56. Primary Containment Isolation						X
57. Closed Systems Isolation Valves						X
60. Control of Release of Radioactive Materials to the Environment		X				
61. Fuel Storage and Handling and Radioactivity Control						X
62. Prevention of Criticality in Fuel Storage and Handling	X					
63. Monitoring Fuel and Waste Storage		X				
64. Monitoring Radioactivity Releases		X				

VIII. Proposed Design Modifications and Issues Requiring Solution for a U.S.-Sited CANDU-PHW Power Plant

In order to facilitate the possible introduction of CANDU-PHW reactors into the U.S., which would require accommodation to the U.S. licensing environment, it is proposed in the following to make a number of design modifications in the current CANDU design. These modifications would consist of, on the one hand, deletion of some safety-related systems, and, on the other hand, extension of other systems. These modifications are proposed not because the current CANDU design is not adequately safe (it certainly is), but in order to accommodate existing differences in Canadian and U.S. licensing requirements as outlined in the foregoing Chapters.

Before listing the proposed modifications, it is useful to first formulate a number of guiding considerations:

(i) It appears that most, if not all, US-NRC licensing criteria, which are not type-specific (i.e., for which the intent is not dependent on the specific characteristics of the nuclear reactor type in question) should be applied across the board for all nuclear power plants in the U.S. Among this class of licensing requirements should certainly be counted those bearing on protection by design against natural phenomena (including seismic events), since there appears to exist no adequate justification for applying in this area different criteria for different reactor types. On this basis a U.S.-sited CANDU-PHW reactor would have to follow U.S. requirements regarding seismic design, including consideration of the OBE and the simultaneous occurrence of a large-scale LOCA and a seismic event of the severity of the SSE.

(ii) There are substantial differences between the safety-related intrinsic characteristics of CANDU-PHW reactors and LWRs. Among the intrinsic characteristics of the CANDU-PHW reactor is the fact that the ECIS is not the final defense against loss of coolable core configuration and core meltdown in case of the large-scale LOCA. It would appear that this fact should allow a certain relaxation of the ECIS requirements for CANDU-PHW reactors with respect to the ECCS requirements in force for LWRs, provided that the back-up defense line (moderator cooling system), is qualified as a safety system in accordance with US criteria (seismically qualified for SSE, ASME-Section III-Class 1 construction, etc.). Such a relaxation of the ECIS requirements for CANDU-PHW reactors with respect to ECCS requirements for LWRs is furthermore justified on the basis that the intent of the ECCS requirements for LWRs is to maintain coolable core configuration, so as to avoid a core-melt accident (Report of the Ergen Committee, 1967). Since the CANDU core and the LWR core differ substantially from each other (the LWR core being relatively closely packed without a dispersed heatsink), a larger degree of fuel damage can be tolerated in the CANDU reactor prior to losing coolable core configuration than in the LWR. It should be stressed here, however, that CANDU reactors do not necessarily suffer a larger degree of core damage than do LWRs following a large-size LOCA; it is solely pointed out that, since maintaining coolable core configuration is the objective, a larger degree of core damage could be tolerated in a CANDU reactor than in a LWR in case of a LOCA.

On the ground of the above considerations the following main design modifications are proposed (see Figure V.1):

(1) Eliminate the Emergency Water Supply (EWS);

(2) Eliminate the Emergency Power Supply (EPS);

(3) Qualify the ECIS to the SSE level;

(4) Qualify the Moderator Cooling System to the SSE level;

(5) Qualify the Shutdown Cooling System to the SSE level;

(6) Qualify the Primary Coolant Make-up System to the SSE level;

(7) Qualify the Service Water System to the SSE level;

(8) Qualify the Main On-Site Diesel Generator Set to the SSE level.

The above proposed modifications, which would make the CANDU-PHW reactor conform in most areas with U.S. licensing requirements, would not constitute a major redesign, since many of the above listed systems are in fact already qualified, for economic reasons, to a higher level of seismicity than is required for safety reasons. (Figure V.1 indicates Canadian seismic requirements; not actual levels of seismic qualification). As can be seen from Table V.6, which gives the actual levels of seismic qualification, to achieve seismic qualification of the above listed systems would in most cases solely require qualification of their power supplies to the SSE level.

It should be noted that the above proposed trade-off, could lead to a slight increase in the CANDU-PHW reactor's sensitivity to common-mode failures relative to the unmodified version of the current CANDU-PHW reactor design: The physical separation of the EPS and EWS from other power and coolant supplies is purposely provided in current CANDU plants to protect against low probability events that could induce common-mode failures (airplane crashes, fires, etc).

Apart from the above proposed modifications, there remain a number of issues which need to be resolved for a U.S.-sited CANDU-PHW power plant. Among the main issues remaining are the following:

(1) Compatibility of the CANDU-PHW reactor's design with the current ASME B&PV Code. Examples of items requiring resolution are:

(a) use of Zr-Nb alloy in the PHTS (for the pressure tubes),

(b) use of rolled joints in the PHTS,

(c) use of special stainless steels in the PHTS (e.g., use of special 403 SS for the endfittings of the pressure tubes),

(d) use of Zircaloy-2 for calandria tubes, control rods guides, and injection nozzles of the second shutdown system (gadolinium injection)

(e) required ASME-Section III-Class (1 or 2) for the calandria;

(2) Required degree of restraining (against pipewhip, etc) of the feeder lines;

(3) Required degree of inservice inspection for the feeders, and the endshields of the calandria.

As is known, the sections of the ASME code dealing with nuclear power plants were to a large extent developed in response to the needs caused by the LWR development. The designers of the CANDU reactor have followed the ASME code to the extent applicable. For some areas, not now covered by the ASME code, special Canadian standards were developed.[22-31] The design solutions followed for the CANDU reactor are technically sound and have been proven in practice to be reliable, not only in Canada, but also in other countries involved in the application of pressure tube reactors. Still, it is not excluded that extension of the ASME Code to cover the required areas could turn out to be one of the critical-path items with respect to introducing CANDU reactors into the U.S., in view of the length of time required for the various Code Cases.

IX. Summary and Conclusions

The Canadian approach to nuclear reactor safety has to a large extent been developed independently. This fact, combined with the fact that CANDU-PHW reactors have a number of intrinsic safety-related characteristics differing substantially from those of other commercial nuclear power reactor types, has led to the development of Canadian licensing criteria that are in some areas different from those developed in the US for LWRs.

The primary issue in evaluating CANDU licensability in the US is therefore not whether CANDU-PHW reactors meet adequate safety standards, but rather how much effort is required to introduce the CANDU technology into the US regulatory environment, as determined by current US-NRC regulatory criteria, guides, procedures, practices, and standards.

The principal differences between the Canadian and US safety approach pertain to the following areas:

(1) Safety Analysis: The Canadian licensing procedures require analysis of dual-failure accidents (i.e., failure of a process system coincident with failure of a safety system), including the occurrence of a maximum-size LOCA simultaneously with unavailability of the ECIS, or impairment of the containment system. A further difference between Canadian and US licensing criteria relative to safety analysis pertains to the maximum permissible conditions for the cladding attained during and following blowdown in case of a postulated LOCA. One of the US requirements is that the cladding temperature shall not exceed 1200°C at any point in the core and at any time during or following the blowdown, whereas the Canadian requirement is more flexible, having the character of a time-at-temperature limit rather than a strict temperature limit. This difference appears, however, to be not essential because Canadian LOCA analyses performed for

current-design CANDU-PHW reactors show the maximum cladding temperatures to be about 1200°C. Furthermore, the underlying intent of the US ECCS criteria is maintaining coolable core configuration, which for CANDU-PHW reactors can be accomplished, even in case of unavailability of the ECIS, by the Moderator Cooling System.

(2) Seismic Design: Consideration of the simultaneous occurrence of a maximum-size LOCA and a maximum-level earthquake is not required in Canada, as is the case in the US. However, Canadian design criteria require consideration of the simultaneous occurrence of a leak in the PHTS, an impairment of the containment system, and the maximum-level earthquake. Consideration of containment impairment is not required in the US. It is noted that the CANDU-PHW reactor is with respect to seismic design in full accordance with the IAEA Codes of Practice and Safety Guides.

(3) Radiation Doses: Current Canadian reference dose limits for single-failure accidents (including the maximum-size LOCA) are smaller than 10 CFR 100 dose limits in the US by factors of 50 and 100, respectively, for whole-body and thyroid exposure. On the other hand, however, a larger degree of conservativeness is incorporated in the US in the determination of the radiological source term for dose calculations than in Canada.

(4) Redundancy: The Canadian approach in many cases is to provide "redundancy by diverse systems" (i.e., diverse systems providing the same safety function) in addition to "redundancy within single systems". On the other hand, Canadian requirements for redundancy of passive components (particularly low-pressure piping) is less stringent than in the US.

(5) Piping Restraints: The Canadian requirements for restraining small-diameter PHTS piping in CANDU-PHW reactors are less stringent than US requirements for LWRs. This is primarily the case for the feeder lines. Canadian analyses indicate, however, that rupture of a feeder will not propagate to other feeders.

In evaluating the safety aspects of CANDU reactors with respect to licensability in the US, it is desirable to clearly distinguish design characteristics that are intrinsic to the Nuclear Steam Supply System from those that are non-intrinsic (i.e., pertaining to subsystems of a more peripheral nature), and that could therefore be modified relatively easily without changing any of the principal characteristics of the NSSS. On the basis of the foregoing evaluations, it is concluded that none of the intrinsic characteristics of CANDU-PHW reactors is expected to constitute a problem with respect to licensability in the US.

Introduction of CANDU-PHW reactors into the US would require some modifications to be made in the current design of peripheral (i.e., of a non-intrinsic nature) systems of CANDU-PHW reactors in order to accommodate differences in licensing requirements in Canada and the US (see Chapter VIII). These proposed modifications bear to a large extent, on differences in Canada and the US concerning the underlying assumptions applied in the seismic design, as well as on differences in methods for providing redundancy. It should be emphasized that these design modifications are proposed solely for the purpose of maintaining consistency within the US licensing environment, in particular with respect to nuclear reactor types already being licensed in the US; these modifications are definitely not proposed out of a need to improve the safety of the current CANDU-PHW design.

A number of issues remain to be resolved if CANDU-PHW reactors are to be introduced into the US. Those, among these issues, that appear to be on the critical path are mainly related to design solutions applied in the CANDU design which are not yet covered by the ASME code. The reason for this is historical: The nuclear sections of the ASME code have, to a large extent, been developed in response to the needs of the LWRs; there was, up to now, no great need in the US for code development in areas of interest to pressure tube reactors, Canada has developed its own codes and standards relative to design aspects not covered by the ASME code.

It is concluded that CANDU-PHW reactors can be introduced into the US, requiring only a relatively minor redesign. The time scale on which this can be accomplished appears to depend not so much on the technical issues involved, but more on economic considerations, and on whether there exists a sufficient interest to do so.

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X. REFERENCES

1. G. C. Laurence, "Power Reactor Siting in Canada", Transactions of the Am. Nucl. Soc., Washington Meeting (November 1968).

2. D. G. Hurst and F. C. Boyd, "Reactor Licensing and Safety Requirements", paper presented at the Annual Meeting of the Canadian Nuclear Association (June 1972).

3. J. H. F. Jennekens, "Recent Developments in Nuclear Plant Licensing in Canada", paper presented at the Annual Meeting of the Canadian Nuclear Association (1974).

4. J. D. Sainsbury, "CANDU-PHW Accident Analysis", paper presented at the Nuclear Power Symposium jointly sponsored by the Canadian Nuclear Association and Atomic Energy of Canada Limited, Montreal (1974).

5. E. M. Yaremy, "CANDU-PHW Reactor and Station Control", paper presented at the Nuclear Power Symposium jointly sponsored by the Canadian Nuclear Association and Atomic Energy of Canada Limited, Montreal (1974).

6. L. Pease, and R. Wilson, "PHWR-Safety", paper presented at the Canadian Nuclear Association 16th Annual International Conference, Toronto (June 1976).

7. J. McCredie, "The Influence of Design and Regulatory Code Changes on a 'Repeat' Project - Pickering G.S. B", paper presented at the Canadian Nuclear Association 16th Annual International Conference, Toronto (June 1976).

8. E. M. Yaremy, "CANDU Reactor Safety Systems", Transactions Am. Nucl. Soc., 23, p. 63, (June 1976). Also existing as full paper.

9. L. Pease and S. Sawai, "Heavy-Water Pressure-Tube Reactor Safety R&D and Effectiveness of Engineered Safety Systems", paper presented at American Nuclear Society Meeting, Washington, D.C. (November 1976).

10. L. Pease and V. G. Snell, "Safety of Heavy Water Reactors", Paper IAEA-CN-36/181 presented at the International Conference on Nuclear Power and Its Fuel Cycle, Salzburg, Austria (May 1977).

11. J. H. Jennekens, "The Regulation of Nuclear Power Stations in Canada", Proceedings of the Am. Nucl. Soc. Meeting on Thermal Reactor Safety, Sun Valley (August 1977).

12. H. S. Irvine, M. Milicev, "CANDU-PHW Risk Assessment", Proceedings of the Am. Nucl. Soc. Meeting on Thermal Reactor Safety, Sun Valley (August 1977).

13. W. T. Hancox, W. C. Harrison, and H. Tamm, "CANDU Reactor Safety Research Program", Proceedings of the Am. Nucl. Soc. Meeting on Thermal Reactor Safety, Sun Valley (August 1977).

REFERENCES, (Contd.)

14. C. G. Duff, "Canadian Summaries", Paper presented at Post-SMIRT International Seminar on Probabilistic and Extreme Load Design of Nuclear Power Plant Facilities, San Francisco (August 1977).

15. W. Evans, P. Ross-Ross, E. LeSurf and H. Thexton, "Metallurgical Properties of Zirconium-Alloy Pressure Tubes and Their Steel Endfittings", IAEA-Paper A/Conf.49/A/159; also Report AECL-3982 (September 1971).

16. W. Langford and L. Mooder, "Metallurgical Properties of Irradiated Cold-Worked Zr-Nb-2.5 Pressure Tubes", Journal of Nuclear Materials 39, pp. 292-302 (1971).

17. AECL Publication (in preparation), "Fracture Behavior of Zirconium-Alloy Pressure Tubes for Canadian Nuclear Power Reactors", International Journal of Pressure Vessels and Piping (1978).

18. P. Ross-Ross, E. Adams, D. Dixon, and R. Metcalfe, "The Performance and Reliability of Primary-Circuit Components in CANDU Reactors", Proceedings of the European Nuclear Conference held at Paris; also Report AECL-5092 (1975).

19. P. Ross-Ross, W. Evans, and W. Langford, "Experience with Zirconium Alloy Pressure Tubes", Paper presented at Third International American Conference on Materials Technology (October 1972).

20. E. C. W. Perryman, "The Pickering Pressure-Tube Crack Experience", Paper presented at BNES Conference, London (November 1977).

21. C. G. Duff (AECL), Private Communication (1977); to be published in C. G. Duff, "Seismic Design Margins for CANDU Nuclear Power Plants", paper at ASME/CSME Joint Conference on Pressure Vessels and Piping with Nuclear and Materials Divisions, June 25-29, 1978, Montreal, Canada.

22. CSA Preliminary Standard N285.1-1975, "General and Construction Requirements for CANDU Nuclear Power Plant Components".

23. CSA Preliminary Standard N285.4-1975, "Periodic Inspection of CANDU Power Plant Components".

24. CSA Standard N287.1-1977, "General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants".

25. CSA Preliminary Standard N287.2-1976, "Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants".

26. CSA Preliminary Standard N287.6-1975, "Pre-Operational Proof and Leakage Rate Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants".

27. CSA Preliminary Standard N287.7-1975, "In-Service Testing and Examination Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants".

REFERENCES, (Contd.)

28. CSA Preliminary Standard Z299.1-1975, "Quality Assurance Program Requirements".

29. CSA Preliminary Standard Z299.2-1975, "Quality Control Program Requirements".

30. CSA Preliminary Standard Z299.3-1975, "Quality Verification Program Requirements".

31. CSA Preliminary Standard Z299.4-1975, "Inspection Program Requirements".

32. J. C. Hesson, et al., "Laboratory Simulations of Cladding-Steam Reactions Following Loss-of-Coolant Accidents in Water-Cooled Power Reactors, Report ANL-7609 (1970).

33. D. M. O. Hobson, "Ductile-Brittle Behaviour of Zircaloy Fuel Cladding", Proceedings of the Meeting on Water Reactor Safety, Salt Lake City (1973).

34. R. E. Pawel, "Oxygen Diffusion in Beta Zircaloy During Steam Oxidation", Journal of Nuclear Material, Vol. 50, p. 247 (1974).

35. C. E. Till, E. M. Bohn, Y. I. Chang, and J. B. van Erp, "A Survey of Considerations Involved in Introducing CANDU Reactors into the United States", Argonne National Laboratory Report ANL-76-32 (January 1977).

36. J. Daniel, A. Molin, et al, "Licensing Assessment of the CANDU-PHW Reactor", Report UE&C-ERDA 770630 (June 1977).