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PRELIMINARY SAFETY AND ENVIRONMENTAL INFORMATION DOCUMENT

VOLUME II

HEAVY-WATER REACTORS

January 1980

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM

U.S. DEPARTMENT OF ENERGY
ASSISTANT SECRETARY FOR NUCLEAR ENERGY
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FOREWORD

The Department of Energy (DOE) Nonproliferation Alternative Systems Assessment Program (NASAP) is a planned program of studies of nuclear power systems, with particular emphasis on identifying and then evaluating alternative nuclear reactor/fuel-cycle systems that have acceptable proliferation-resistance characteristics and that offer practical deployment possibilities domestically and internationally. The NASAP was initiated in 1977, in response to President Carter's April 1977 Nuclear Power Policy Statement.

The NASAP objectives are to (1) identify nuclear systems with high proliferation resistance and commercial potential, (2) identify institutional arrangements to increase proliferation resistance, (3) develop strategies to implement the most promising alternatives, and (4) provide technical support for U.S. participation in the International Nuclear Fuel Cycle Evaluation (INFCE) Program.

The NASAP is not an assessment of all future energy-producing alternatives. Rather, it is an attempt to examine comprehensively existing and potentially available nuclear power systems, thus providing a broader basis for selecting among alternative systems. The assessment and evaluation of the most promising reactor/fuel-cycle systems will consider the following factors: (1) proliferation resistance, (2) resource utilization, (3) economics, (4) technical status and development needs, (5) commercial feasibility and deployment, and (6) environmental impacts, safety, and licensing.

The U.S. Department of Energy (DOE) is coordinating the NASAP activities with the U.S. Nuclear Regulatory Commission (NRC) to insure that their views are adequately considered at an early stage of the planning. In particular, the NRC is being asked to review and identify licensing issues on systems under serious consideration for future research, development and demonstration. The Preliminary Safety and Environmental Information Document (PSEID) is the vehicle by which NASAP will provide information to the NRC for its independent assessment. The PSEID contains the safety and environmental assessments of the principal systems. Special safeguards measures will be considered for fuel cycles that use uranium enriched in U-235 to 20% or more, uranium containing U-233 in concentrations of 12% or more, or plutonium. These measures will include the addition of radioactivity to the fuel materials (i.e. spiking), the use of radioactive sleeves in the fresh fuel shipping casks, and other measures. The basis for the safeguards review by NRC is contained in Appendix A.

The information contained in this PSEID is an overlay of the present safety, environmental, and licensing efforts currently being prepared as part of the NASAP. It is based on new material generated within the NASAP and other reference material to the extent that it exists. The intent of this assessment is to discern and highlight on a consistent basis any safety or environmental issues of the alternative systems that are different from a reference LWR once-through case and may affect their licensing. When issues exist, this document briefly describes research, development, and demonstration requirements that would help resolve them within the normal engineering development of a reactor/fuel-cycle system.

The preparation of this document takes into consideration NRC responses to the DOE preliminary safety and environmental submittal of August 1978. Responses to these initial comments have been, to the extent possible, incorporated into the text. Comments by the NRC on this PSEID were received in mid-August 1979 and, as a result of these comments, some changes were made to this document. Additional comments and responses were incorporated as Appendix B. Comments that are beyond the scope and resources of the NASAP may be addressed in research, development, and demonstration programs on systems selected for additional study. The intent of this document (and the referenced material) is to provide sufficient information on each system so that the NRC can independently ascertain whether the concept is fundamentally licensable.

This PSEID was prepared for the DOE through the cooperative efforts of the Argonne National Laboratory, the Oak Ridge National Laboratory, and NUS Corporation.

The principal sources of information used for the preparation of this document are as follows:

Chapter 1.0, General Description, and the associated figures and tables are based on the "Combustion Engineering Study on Heavy Water Reactors," by N. Shapiro, J. F. Jesick, et al.

Section 2.1, Fuel Cycle Description, and the tables on charge/discharge data and mass-flow diagrams are based on "NASAP/INFCE Reactor Mass Flow Data Base," December 1978, and adapted by the Argonne National Laboratory and NUS Corporation to conform with the format of this document.

Sections 2.2, Safety Considerations, and 2.4, Licensing Status and Considerations are based primarily on a report titled Preliminary Evaluation of Licensing Issues Associated with US-Sited CANDU-PHW Nuclear Power Plants, ANL-77-97, by J. B. van Erp, and a paper titled "Licensing Evaluation of CANDU-PHW Nuclear Power Plants with Respect to U.S. Regulatory Requirements," presented at the ENS-ANS International Topical Meeting on Nuclear Power Reactor Safety at Brussels, Belgium, October 1978, by the same author. Other sources are as noted in the text.

Section 2.3 is based on engineering judgment and calculations on the part of the NUS Corporation staff.

Section 2.5 is a joint effort of the Argonne National Laboratory and the staff of Combustion Engineering, Inc.

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GENERAL DESCRIPTION

1.1 PLANT DESCRIPTION

The conceptual heavy-water-reactor (HWR) plant described here is based on the Canadian pressurized-heavy-water deuterium/uranium reactor (CANDU-PHW) and, in particular, on the standard CANDU 600 reactor currently under construction at the Gentilly II and Point Lepreau stations in Quebec and New Brunswick, respectively. The conceptual design differs from the Canadian CANDU 600 design by an increase in core heat output to 3,800 MWt, the use of slightly enriched uranium fuel, modifications to increase station efficiency, and design changes made to facilitate conformance to U.S. design or licensing practices. Where possible, the design and features of the CANDU-PHW reactor have been retained to take advantage of the extensive development and proved performance of the Canadian HWR design.

A detailed description of the HWR design, including the various subsystems and components, is given in Reference 1. Tables 1-1 and 1-2 present the reactor performance and design specifications.

The layout for the single-unit conceptual HWR plant is shown in Figure 1-1. The arrangement of buildings provides for the optimum use of space for the various functions performed in the plant and provides controlled access to vital areas. Cooling towers are provided to reject heat from the condenser and from the moderator heat exchanger. The arrangement of safety-related structures is shown in Figures 1-2 through 1-9.

The reactor containment building (RCB) houses the nuclear steam supply system (NSSS). It is an earthquake-resistant (seismic Category I) reinforced-concrete cylindrical structure with a hemispherical dome and a flat reinforced-concrete base. The reinforced-concrete base is 165 feet in diameter and 10 feet thick; it is founded 34 feet below finished grade. The upright cylindrical portion of the containment has an inside diameter of 146 feet and measures 183 feet 6 inches from the top of the foundation mat to the springline of the dome; the wall is 4 feet 6 inches thick. The hemispherical dome is of reinforced concrete, has an inside height of 73 feet, and is 3 feet 6 inches thick. The inside height from the top of the mat to the inside of the dome is 256 feet 6 inches. A 3/8-inch-thick carbon-steel liner is applied to the inner surface of the cylinder, a 1/2-inch-thick liner on the inside of the dome, and a 1/4-inch-thick liner on the top of the foundation mat. A 100-ton polar crane for construction and maintenance operations is located near the springline of the containment and is supported from the wall of the containment.

The reactor service building (RSB) houses all the auxiliary systems, emergency systems, fuel-handling equipment and related systems, and all systems associated with heavy-water inventory control. It is a seismic Category I building located north-northwest of the reactor containment building, with reinforced-concrete exterior walls, interior walls, floor slabs, and roof slab. The exterior walls are at least 2 feet thick. The spent-fuel pool is 25 feet square and will accommodate a 1-year accumulation of spent fuel plus one core of fuel. It is lined with 1/4-inch-thick type 304 austenitic stainless steel plate, which provides a watertight membrane.

The control, auxiliary, and diesel-generator building (CADB) houses the control and electrical equipment, including support systems required for plant operation. It also houses such essential features as the main control room, remote shutdown area, and the emergency diesel generators. It is a seven-story, reinforced-concrete seismic Category I structure, 124 feet wide, 113 feet long, and 91 feet 6 inches high. Two separate and independent cable-spreading rooms, one above and one below the control room, are provided for power, control, and instrumentation channels. In addition, a network of four separate and independent vertical cable chases and containment-penetration rooms is included.

A containment annulus building (CAB) is provided between the CADB and the containment. This building subtends an arc of 102 degrees, is 16 feet wide, extends the full height of the CADB, and is supported on a common mat with the CADB. Steel structural support is provided for routing the various cable trays to the outside of the containment. The building is designed to meet the requirements of a seismic Category I structure. The exterior walls, interior walls, and roof slab are reinforced concrete. The exterior walls are at least 2 feet thick. Reinforced-concrete walls separate the diesel generators and support the upper floors.

The penetrations building houses the main steam and feedwater piping and valves, and portions of the moderator system. The building is a seismic Category I structure of reinforced concrete. It is on a common foundation mat with the containment and shares a common wall with the containment on one side; on the other three sides and on the roof it has 2-foot-thick walls. The building is 120 feet long, 32 feet wide from the outside of the containment wall to the edge of the turbine building, and 98 feet high. The main steam and boiler feedwater lines are separated from the moderator-system components by reinforced-concrete walls. The area containing the main steam piping includes the power-operated relief valves and safety relief valves. The relief valves are provided with vent stacks for steam discharge to the environment. Boiler feedwater piping valves are also in this building.

Some of the structures included in the plant complex are not designed to meet the requirements of seismic Category I structures. These contain non-safety-related equipment and components that are not required for safety or for safe shutdown. These structures are normally constructed of structural steel framing, metal siding, and concrete-plank roofing; they are founded on concrete spread footings. An example is the turbine building, which houses the turbine generator, condensers and associated equipment, feedwater heaters and pumps, and auxiliary equipment. Seismic Category I components or equipment are not housed in this building (see Figures 1-10 and 1-11). Other buildings in this category are the following:

1. Administration building and warehouse
2. Security building
3. Circulating-water and service-water pumphouse
4. Makeup-water intake structure
5. Makeup-water pretreatment building
6. Cooling-tower switchgear building
7. Fire-water pumphouse

The nuclear steam supply system is housed in the containment building and consists of the reactor assembly (including the fuel), the heat-transport system, steam generators, primary-system pumps, pressurizer, shutdown and control systems, and instrumentation. The horizontal-pressure-tube pressurized reactor is moderated and cooled with heavy water and fueled with slightly enriched uranium dioxide. A simplified diagram showing the features of this type of reactor is presented in Figure 1-12. The reactor is installed in a steel-lined concrete vault filled with light water for shielding. The main reactor structure is an austenitic stainless steel calandria with integral end shields and a peripheral internal thermal shield. The calandria contains the low-temperature (200°F) heavy-water moderator and operates at near-atmospheric pressures. The horizontal coolant tubes, in which the fuel resides, are located within tubes passing through the calandria. These calandria tubes are made of thin-walled (1.4 mm minimum wall thickness) Zircaloy and are separated from the coolant (or pressure) tubes by a sealed annulus containing dry nitrogen. The purpose of this annular gas gap is to provide thermal insulation between the high-temperature tubes and the low-temperature heavy-water moderator. The coolant tubes are located within the calandria tubes and are supported in sliding bearings at the end shields of the calandria. Heat produced in the fuel by fission is removed by pressurized heavy water flowing past the fuel-element bundles. In heavy-water reactors of this type, these coolant tubes serve as the pressure boundary between the high-pressure coolant and the moderator, which is operated at near-atmospheric pressure. They are made of a zirconium-2.5% niobium alloy and are approximately 6 meters long and 10.3 cm inside diameter. The conceptual HWR plant is designed to operate at a somewhat higher primary-system pressure than the CANDU 600 (2,250 vs. 1,600 psi for the CANDU 600) to allow primary-system coolant temperatures and station electrical efficiency to be increased. In order to accommodate the higher system pressure, the thickness of the coolant tubes has been increased from 4.34 mm in the CANDU 600 design to 5.79 mm; other than this increase in coolant-tube thickness, the design features of the standard CANDU 600 coolant tube have been retained.

Fuel for the reactor is in the form of cylindrical bundles, approximately 0.5 meter long. Each long bundle consists of 37 fuel elements, each of which in turn consists of approximately 29 fuel pellets stacked end to end and sealed in a Zircaloy-4 sheath. The fuel elements are attached mechanically at their ends to form a cylinder approximately 100 mm in diameter, with a small space being maintained between elements by spacers attached to the element cladding. Twelve of these fuel bundles are stacked end to end in each fuel channel.

The heat-transport system is shown schematically in Figure 1-13. The main heat-transport system circulates pressurized heavy water through the fuel channels to remove heat produced by the fission process. The heat is carried to the steam generators, in which it is transferred to light water to form the steam that subsequently drives the turbine generator. The simplified flow diagram of Figure 1-13 shows a single primary heat-transfer loop. Each loop consists of two steam generators and two primary-coolant pumps at each end of the reactor, so that flow is in one direction through half the fuel channels and in the opposite direction through the other half. In the conceptual HWR design, two such circulation loops are provided, each serving one-half of the reactor core. In addition to the main heat-transport system, a separate system is provided to maintain the temperature of the heavy-water moderator. Because of the low temperature of the moderator, heat extracted by the moderator cooling system is not used to generate electricity; it is dissipated directly by the heat-rejection system.

As with all pressurized-water-reactor (PWR) concepts, steam generators are used to transfer heat from the reactor coolant and to boil light water contained in a secondary circuit. Steam generated in the steam generators is then used to drive the turbine generator. Four identical steam generators consisting of inverted vertical U-tube bundles installed within a shell and containing integral preheaters are provided; these steam generators are identical in concept with those currently used in pressurized HWRs and LWRs, differing only in dimensions.

In normal operation, the main method for reactivity control is on-power refueling. (For first cores, control by soluble boron moderator poison is also used until the excess reactivity of the initial core is depleted.) With on-power refueling, the reactor operates with essentially no excess reactivity (except for that contained in the normally inserted adjuster rods). Fresh-fuel bundles are charged into the fuel channel by a remotely operated fueling machine, while spent-fuel bundles are simultaneously discharged by another refueling machine at the opposite end of the reactor. With on-power refueling, fuel is continually charged and discharged so as not to require reactivity-control systems to compensate for fuel depletion. A light-water zonal control system and vertical absorber rods are provided to control the small reactivity changes that occur between refuelings and to adjust the local power distribution within the reactor core. Vertical adjuster rods (control rods containing a graded vertical loading of cobalt poison) are also provided for xenon power override. These adjuster rods are normally fully inserted during operation and are withdrawn after shutdown or operation at low power to increase reactor power by providing the reactivity to override the buildup of xenon.

Two diverse reactor-shutdown systems are provided. The first system consists of control rods that are released on trip to fall into the core by gravity. The second shutdown system consists of equipment to inject gadolinium poison into the moderator.

In-core instrumentation is provided to allow automatic control of reactor power and flux shape and to monitor local core behavior. Central to the instrumentation and control system are large-capacity digital computers for station control, alarm annunciation, and data display. The computer system also serves to coordinate reactor power level with turbine demand, to manipulate the various reactivity-control devices so as to maintain a desired flux shape, to initiate slow and fast power cutbacks to keep the plant parameters within limits that will avoid a reactor trip, and to control other plant parameters, such as steam-generator and pressurizer pressure and water levels.

The major modifications relative to the CANDU reactor are described in Section 1.2.

Table 1-1. Generalized performance specifications: heavy-water reactor fueled with slightly enriched uranium[a]

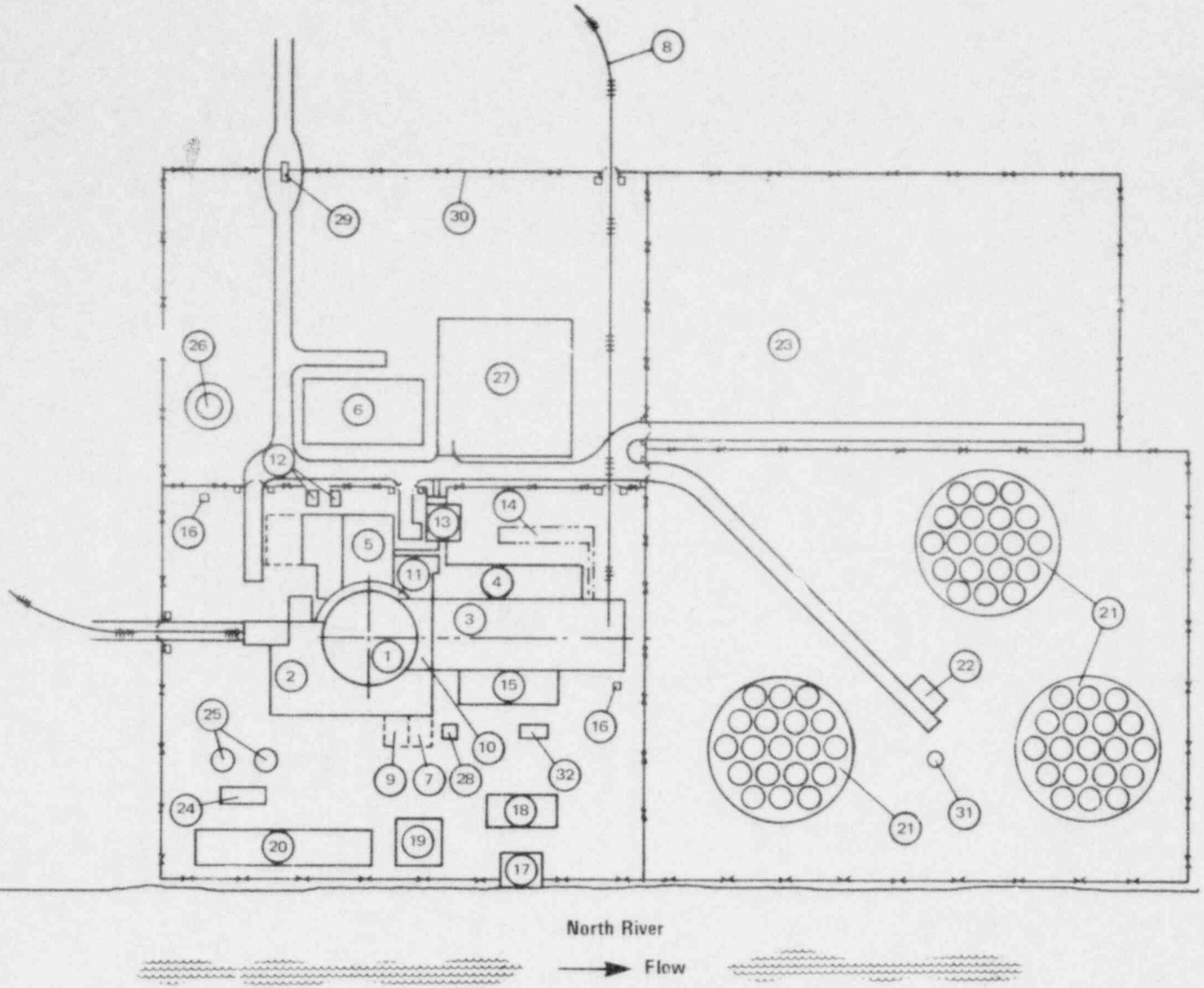
Power plant performance parameters	
Reactor thermal power output, MW	4,029
Electrical power output, MWe	
Gross	1,343
Net	1,260
Plant heat rate, Btu/kW-hr	10,913
Core design and performance parameters	
Core heat output, MWt	3,800
Core volume, liters	359,000
Core loading, kg	
Heavy metal	166,056
Fissile fuel (first core natural uranium)	1,181
Conversion ratio	0.72
Average discharge burnup, MWd/MTHM[b]	19,750
Peak discharge burnup, MWd/MTHM[b]	23,000
Fuel type	UO_2
Reactor inlet temperature, °F	570
Reactor outlet temperature, °F	639
End-of-cycle excess reactivity	0

[a]Natural uranium first core, 1.2 wt% reload fuel enrichment, once through.

[b]Heavy metal charged.

Table 1-2. Reactor design specifications

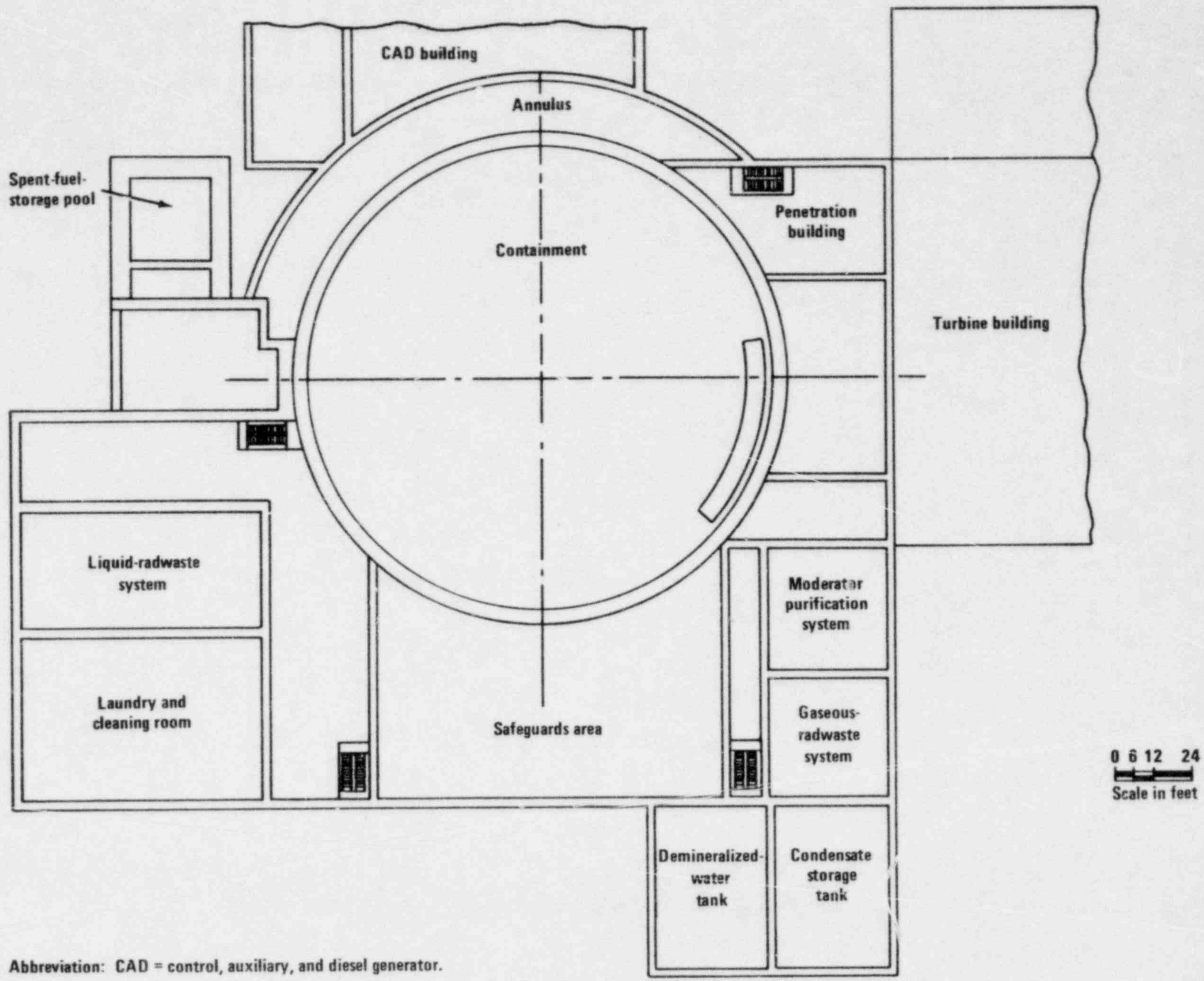
Geometrical information	
Core height, cm	594.4
Number of core enrichment zones	1
Number of assemblies	8,880
Equivalent diameters, cm	877.3
Number of rods, per assembly	37
Rod pitch-to-diameter ratio, minimum	1.20
Overall assembly length, cm	49.53
Lattice pitch, cm	28.575
Assembly material	Zircaloy-4
Cladding parameters	
Cladding outside diameter, mils	515
Cladding wall thickness, mils	16.5
Cladding material	Zircaloy-4
Specific power, kW/kg fissile material	3,412
Power density, kW/kg HM	24.3



PLANT NOMENCLATURE

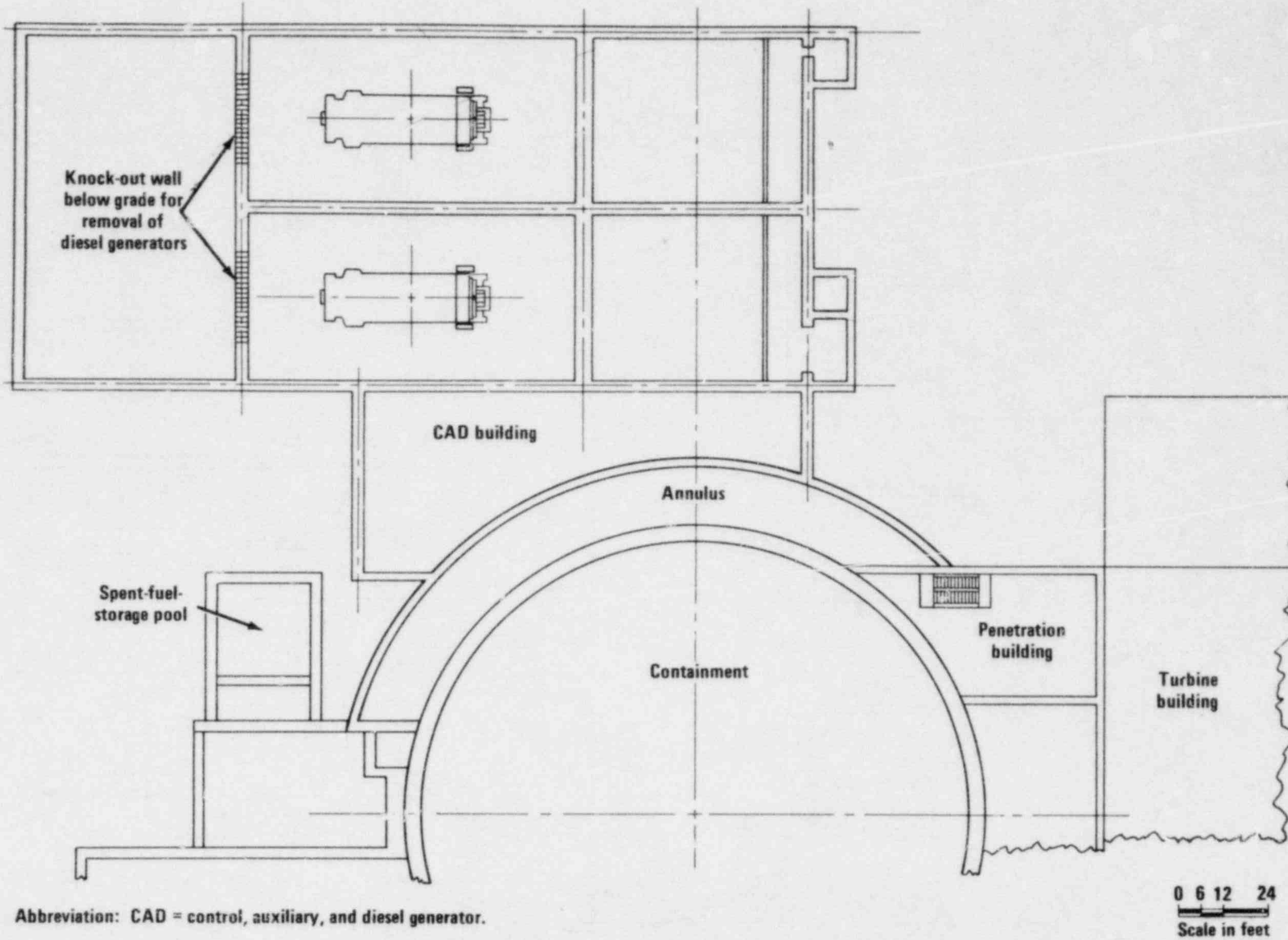
1. Reactor containment building
2. Reactor service and fuel-handling building
3. Turbine building
4. Heater bay
5. Control and diesel building
6. Administration building
7. Condensate-storage tank
8. Railroad
9. Demineralized-water-storage tank
10. Penetration building
11. Annulus building
12. Diesel storage tanks
13. Security building
14. Transformer area
15. Circulating water and service pump building
16. Air-intake structure
17. Makeup-water intake structure
18. Makeup-water pretreatment building
19. Holding pond plant effluent
20. Ultimate heat sink cooling-tower structure
21. Cooling towers
22. Cooling-towers-switchgear building
23. Switchyard
24. Fire-water pump house
25. Fire-water-storage tanks
26. Fuel-oil-storage tank
27. Parking area
28. Heavy-water upgrading tower
29. Guardhouse
30. Security fence
31. Anhydrous-ammonia storage tank
32. Ammonia-water storage tank

Figure 1-1. Plot plan.



Abbreviation: CAD = control, auxiliary, and diesel generator.

Figure 1-2. General arrangement of safety-related structures; plan, elevation (-) 21 ft-3 in.



Abbreviation: CAD = control, auxiliary, and diesel generator.

Figure 1-3. General arrangement of safety-related structures; plan, elevation (-) 17 ft-3 in.

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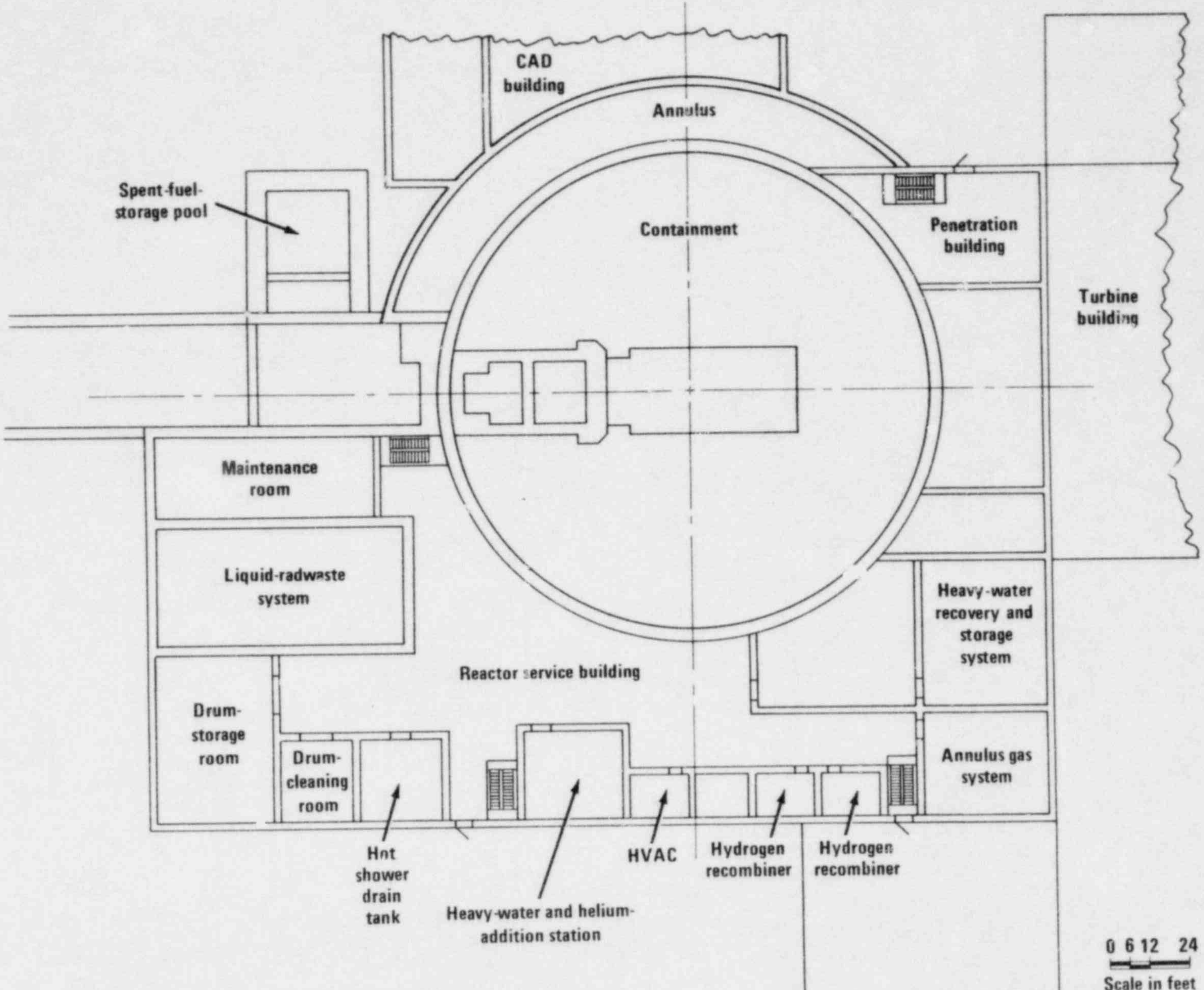

Abbreviation: CAD = control, auxiliary, and diesel generator.

Figure 1-4. General arrangement of safety-related structures; plan, elevation 0.

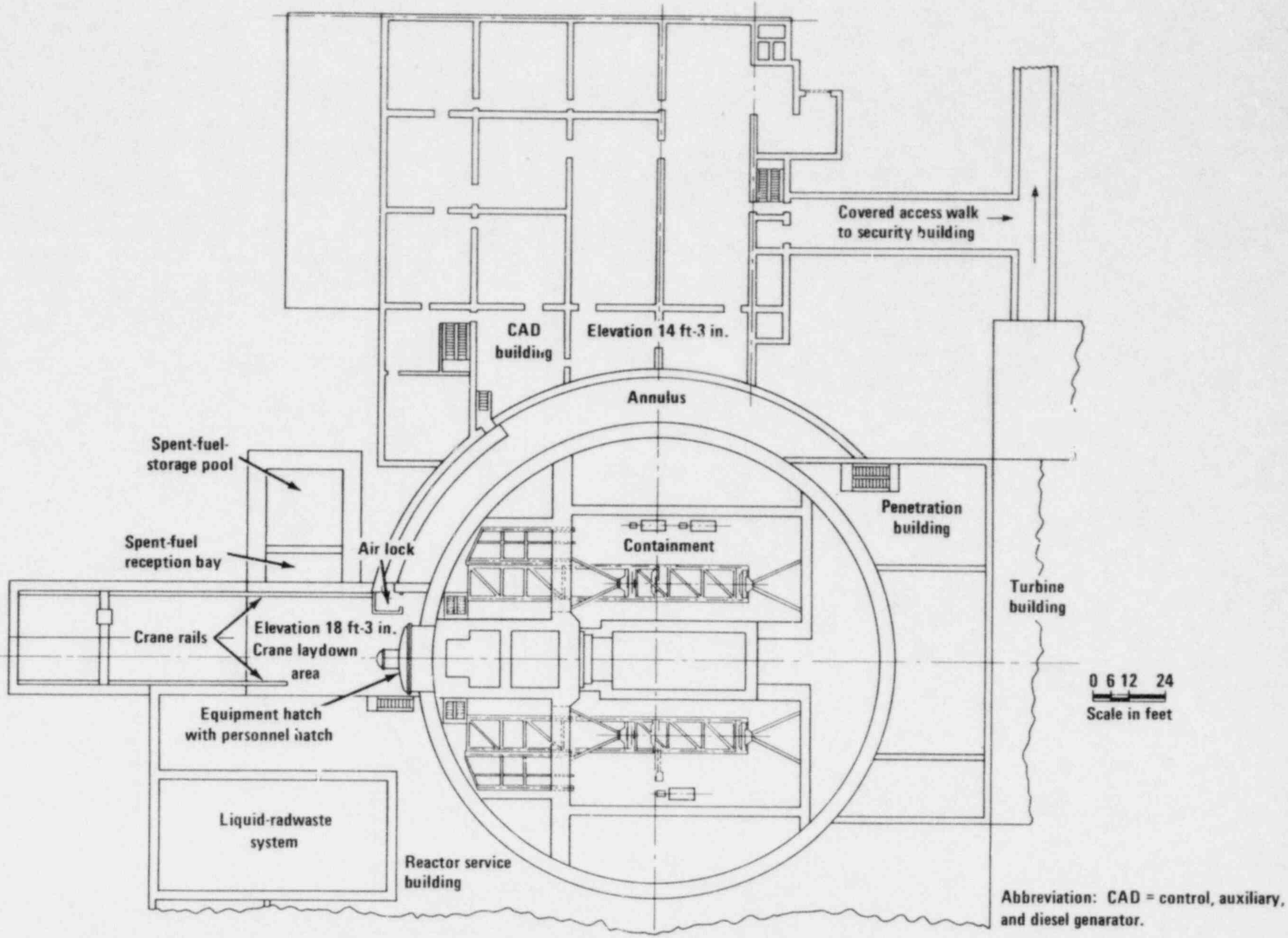

Figure 1-5. General arrangement of safety-related structures; plan, elevation 14 ft-3 in.

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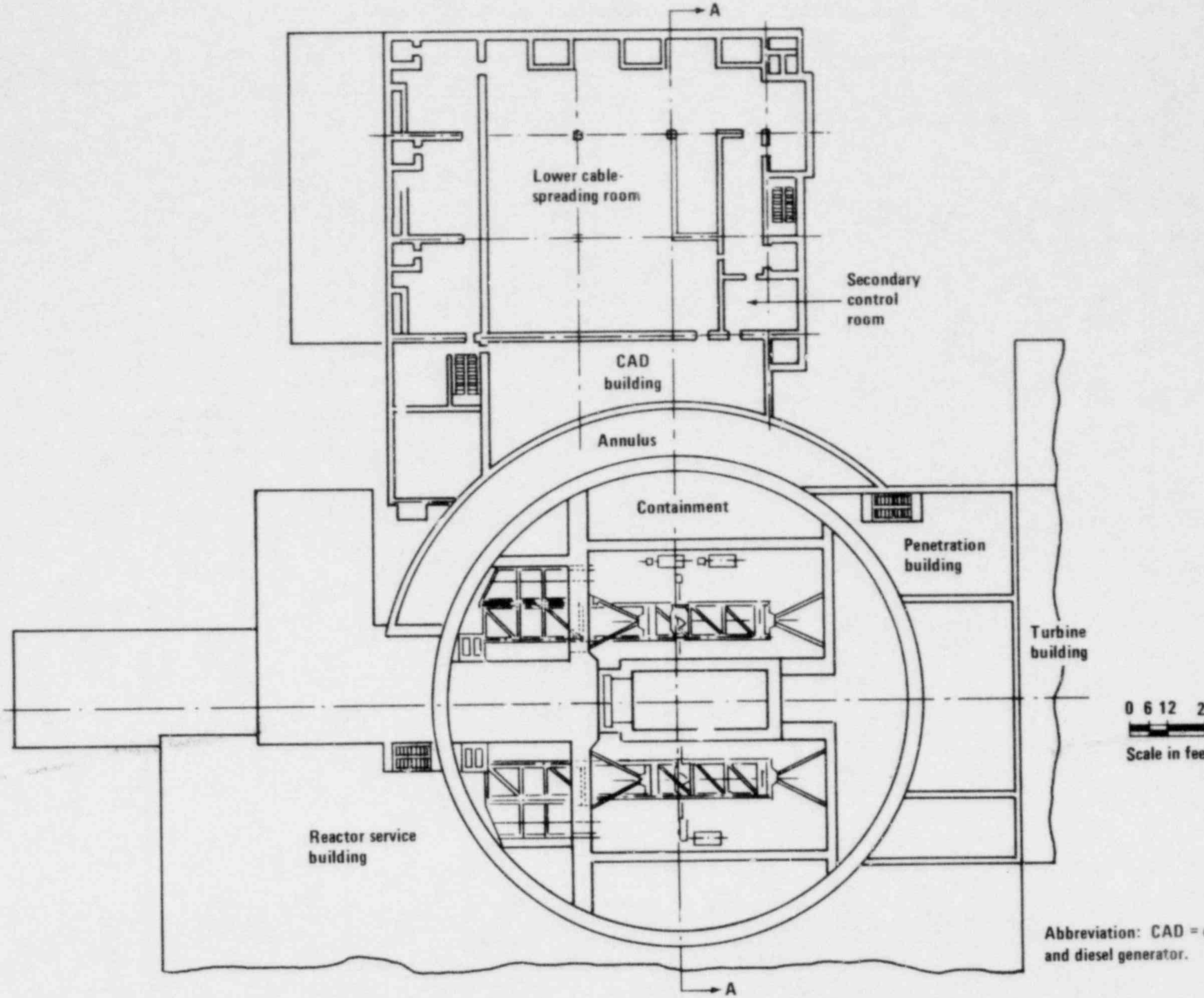

Abbreviation: CAD = control, auxiliary, and diesel generator.

Figure 1-6. General arrangement of safety-related structures; plan, elevation 37 ft-9 in.

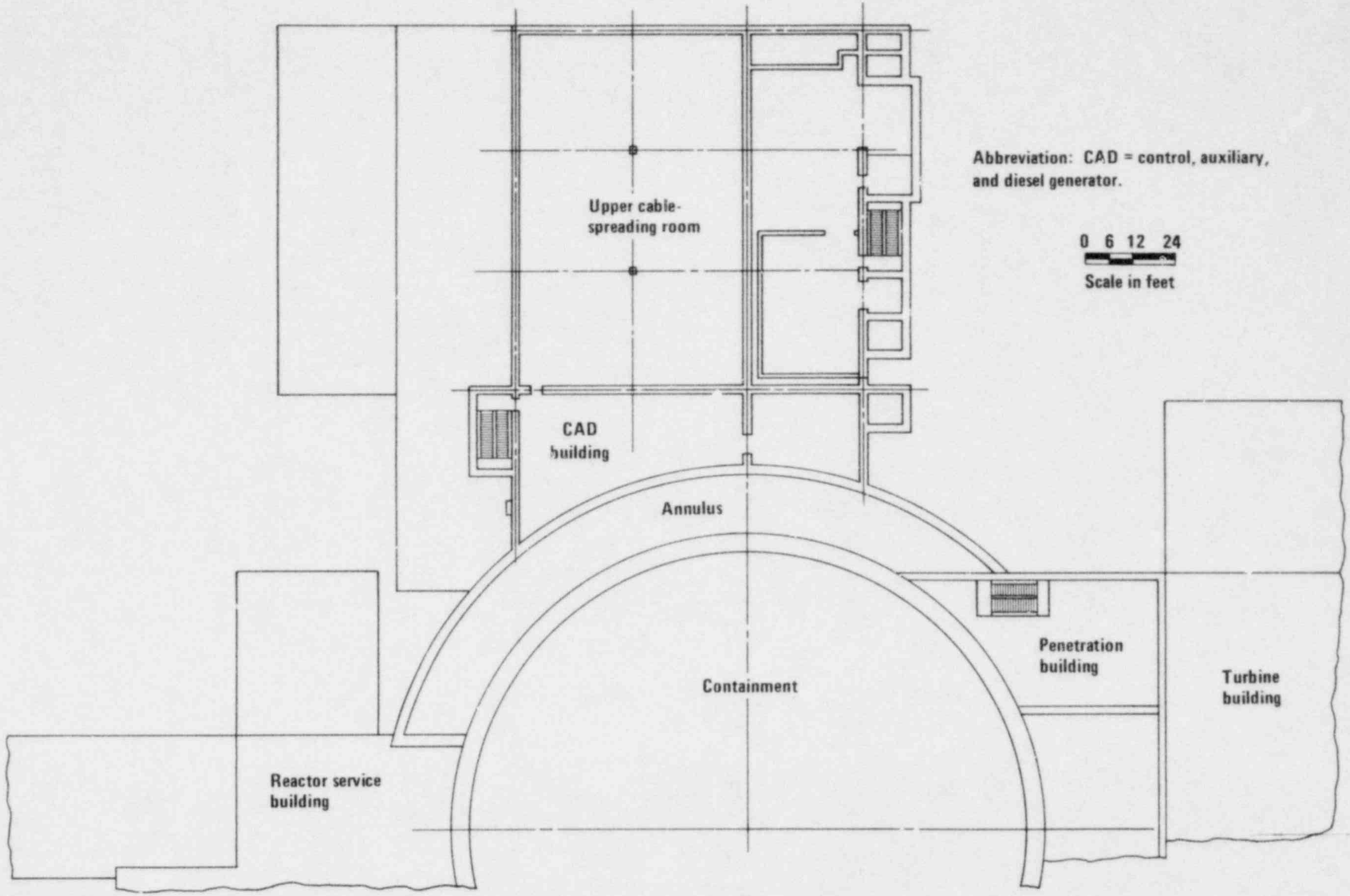

Figure 1-8. General arrangement of safety-related structures; elevation 79 ft-3 in.

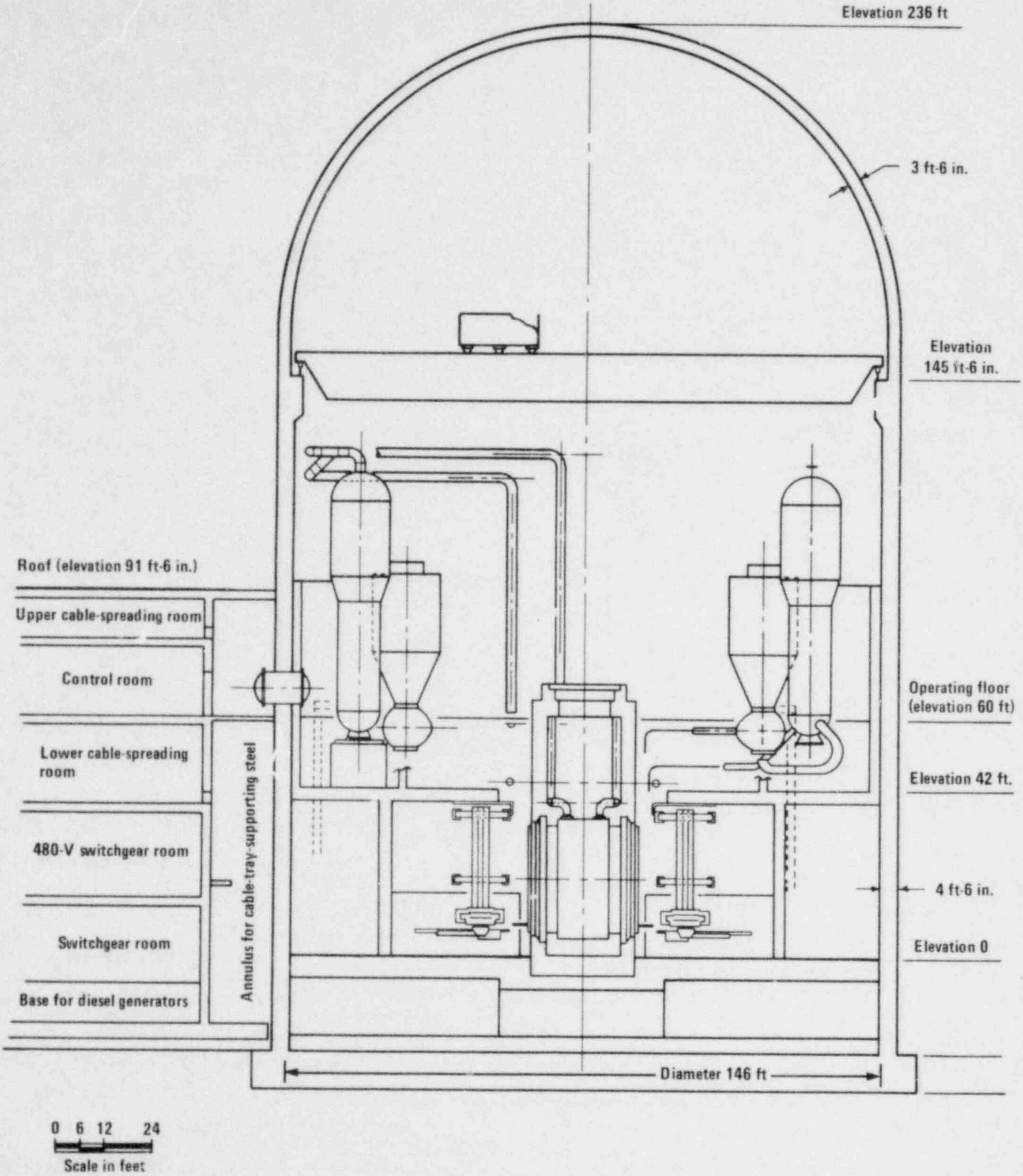

Figure 1-9. General arrangement of safety-related structures; section A-A.

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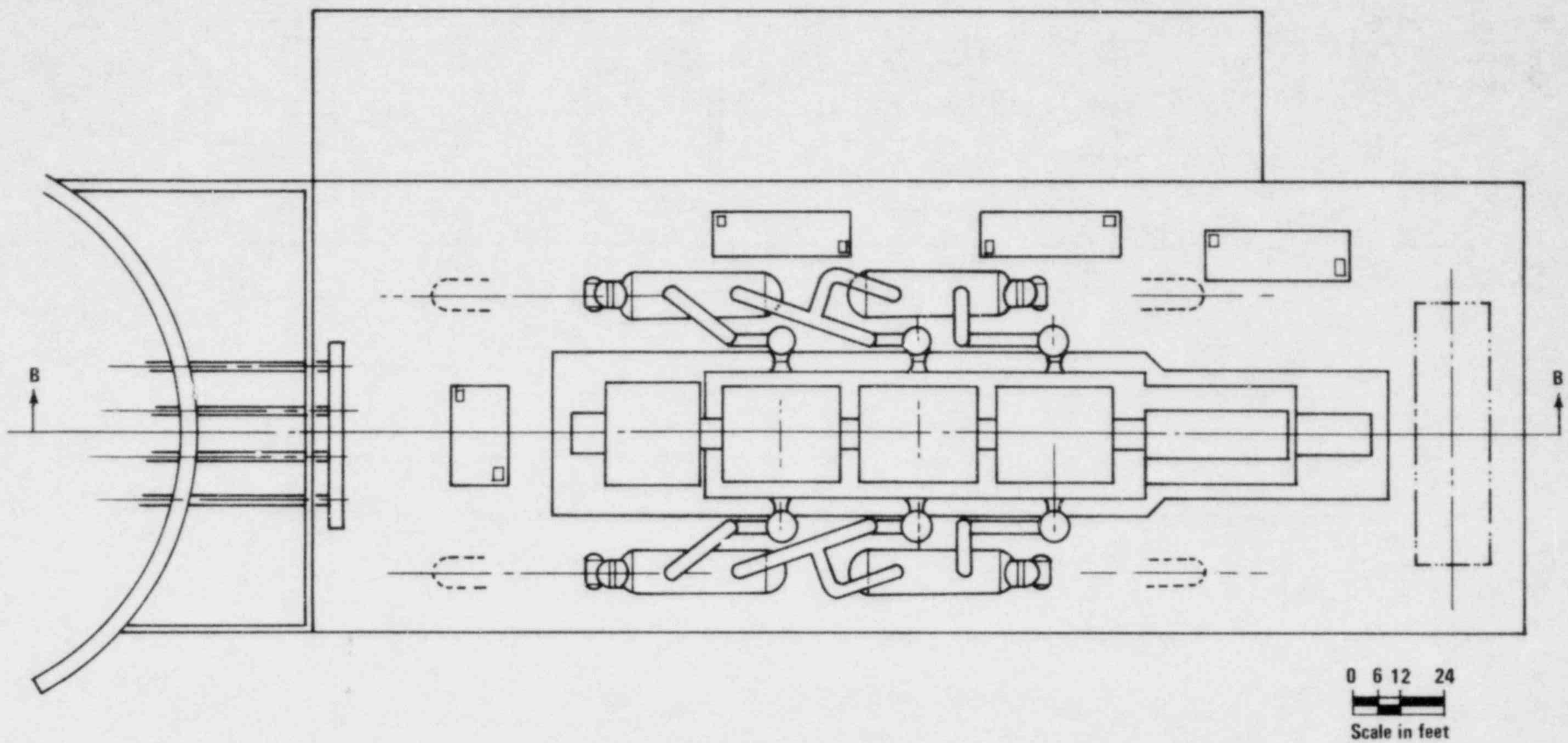


Figure 1-10. Turbine-building general arrangement; plan, elevation 55 ft.

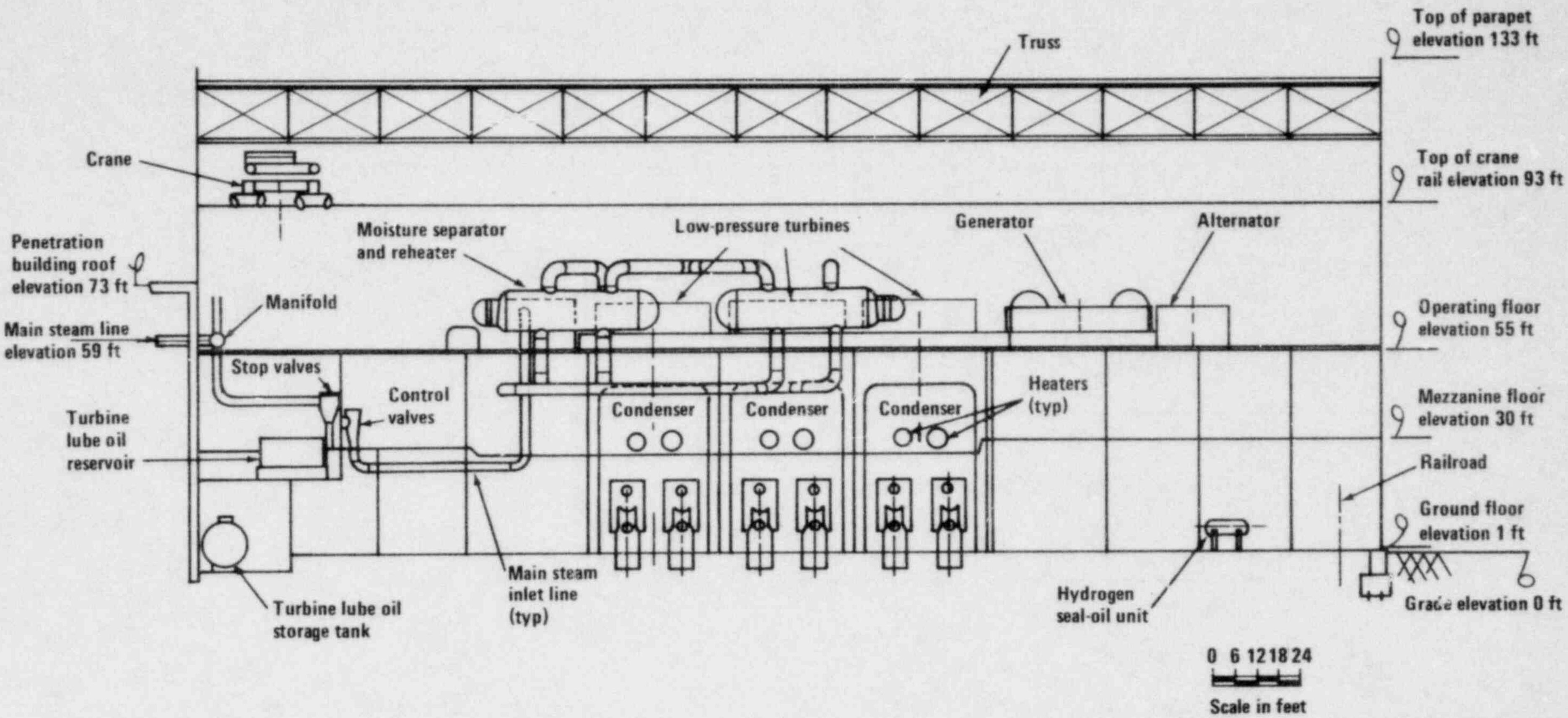
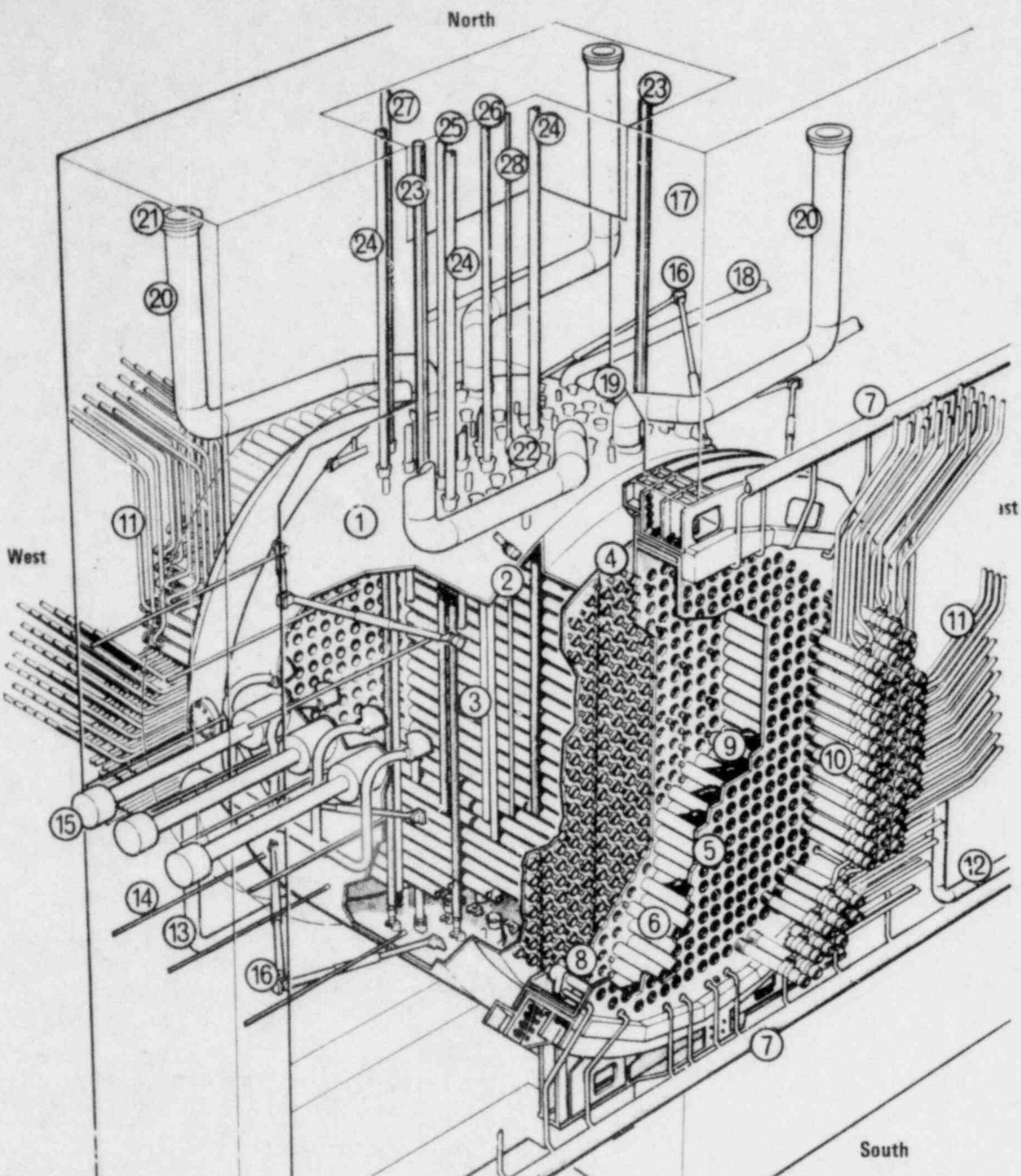


Figure 1-11. Turbine building, section B-B.



1. Calandria
2. Calandria shell
3. Calandria tubes
4. Baffle plate
5. Fueling 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 monitor and poison injection
15. Ion chamber
16. Earthquake restraints
17. Vault wall
18. Vault cooling pipes
19. Moderator overflow
20. Pressure relief pipes
21. Pressure relief discs
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 monitor

Figure 1-12. Reactor assembly.

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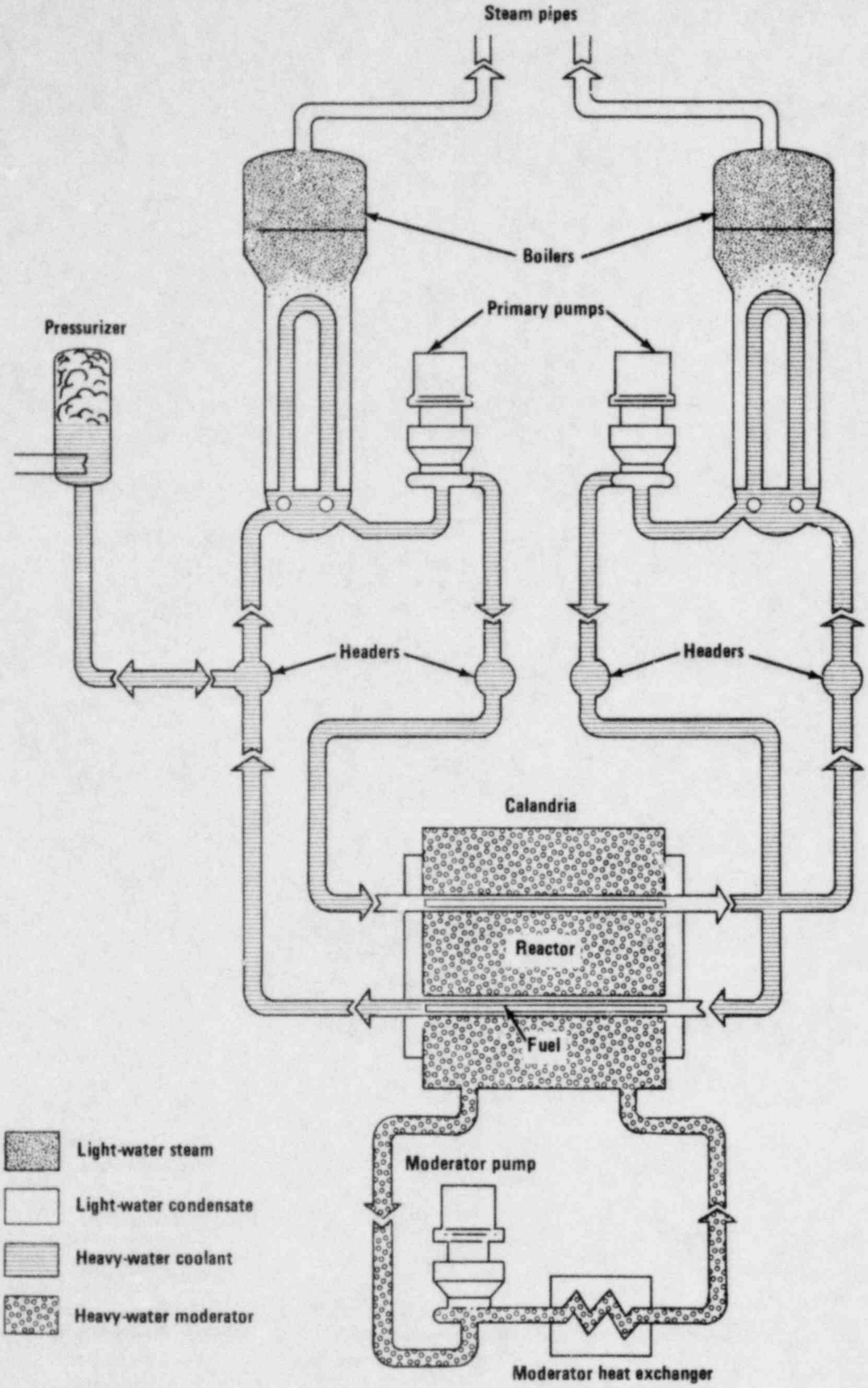


Figure 1-13. Simplified flow diagram of heavy-water reactor.

1.2 MODIFICATIONS RELATIVE TO THE CANDU REACTOR

As mentioned in Section 1.1, the conceptual HWR design described in this report is based on the Canadian pressurized-heavy-water deuterium/uranium reactor (CANDU-PHW) and, in particular, the standard CANDU 600 reactors (Refs. 2 and 3) currently under construction at the Gentilly II and Point Lepreau stations in Quebec and New Brunswick, respectively.

One of the objectives established for the conceptual HWR design was to minimize changes from existing CANDU-PHW designs so as to enhance the prospects for early deployment and to minimize research and development requirements. As a result, design modifications were restricted to those that fit into the following categories:

1. Modifications that would result in significant improvements in fuel utilization and reductions in capital costs
2. Modifications resulting from consideration of current U.S. licensing requirements, construction practices, and sites.

The major changes incorporated into the conceptual HWR design with respect to the standard CANDU 600 reactor are as follows:

1. A net electrical output of 1,260 MWe, versus 600 MWe for the standard CANDU design.
2. The use of slightly enriched (1.2 wt%) uranium fuel rather than natural uranium.
3. Primary-system coolant pressure of 2,250 psi rather than 1,600 psi, to increase the net electrical efficiency from 29 to 31.3%. To accommodate this increase in primary-system pressure, the coolant-tube wall thickness has been increased from 4.34 to 5.79 mm.
4. A heavy-water reflector thickness of 33 cm, about half the thickness used in the standard CANDU design, to reduce the heavy-water inventory.
5. A two-loop primary-system configuration incorporating four steam generators and four primary coolant pumps, in contrast to the three-loop configuration typically used for large CANDU plants. The use of the two-loop configuration with larger steam generators allows containment size and equipment costs to be reduced, thereby reducing plant capital costs.

Other design modifications are described in the sections that follow.

1.2.1 CONTAINMENT

The containment was modified from the CANDU design to a U.S.-type containment structure incorporating the following features:

1. Elimination of the dousing tank located near the top of the containment shell and the addition of an emergency core-cooling system of the type used in U.S. PWRs to facilitate licensing in the United States.
2. Replacement of epoxy coating in the containment with a carbon-steel liner. The carbon-steel liner can be provided with test channels to insure the maintenance of a vapor-tight containment at all times. A metal liner permits complete sealing of containment penetrations to the liner, thus providing a homogeneous liner material that can be fully inspected for flaws and pinhole leaks.

3. Seismic design for a safe-shutdown earthquake of 0.25g and an operating-basis earthquake of 0.125g.

1.2.2 REACTOR SERVICE AND FUEL-HANDLING BUILDING

This building was redesigned to reduce its size. Some of the functions performed in the CANDU design were transferred to other structures. The following major changes were incorporated:

1. The building is designed as a seismic Category I reinforced-concrete structure rather than as a structural steel structure with siding.
2. The size of the spent-fuel storage pool has been reduced to four-thirds of a core (the conventional spent-fuel storage pool can accommodate all the spent fuel produced in 10 years of operation).
3. The control room is relocated in the control, auxiliary, and diesel-generator building.
4. A loading area is provided for all equipment and supplies entering or leaving the safety-related structures of the power plant. This is a controlled-access area.
5. Equipment and components for the engineered safeguards systems are contained in this building.

1.2.3 CONTROL, AUXILIARY, AND DIESEL-GENERATOR BUILDING

This structure is new to the CANDU design. Functions from various locations in the CANDU system were combined to provide an arrangement that would meet NRC criteria. One of the features included in this design is the provision of redundant emergency diesel generators in the lowest level of the CAD building. Each diesel generator provides 100% of the electricity requirements for safety-related systems that must operate during and after an abnormal plant condition.

The CAD building also houses a control room provided with life-support systems, including redundant air intakes, to enable operators to perform their duties under abnormal plant operating conditions. A secondary control room is located in the CAD building to provide a backup means of controlling the reactor systems in the unlikely event that the main control room becomes uninhabitable.

Redundant cable-spreading rooms, one above and one below the control room, are provided. Fire-detection and fire-fighting systems are located in these rooms to insure immediate response in the unlikely event of a fire.

Other features of the new design are as follows:

1. Major switchgear is located in the CAD building.
2. The structure is designed to seismic Category I requirements.
3. Security and control of personnel within the reactor plant are maintained from the CAD building.

1.2.4 PENETRATIONS BUILDING

The penetrations building is new to the CANDU system. It is a seismic Category I reinforced-concrete structure designed to--

1. Protect the main steam and boiler-feedwater containment-isolation valves

2. House the moderator-cooling heat exchangers and moderator-circulating pumps (previously located in the containment)
3. Contain and support the moderator-cooling heat-exchanger cooling system (an ammonia absorption system)

1.2.5 ANNULUS BUILDING

The annulus building is located between the containment and the CAD building. This seismic Category I reinforced-concrete structure is new to the CANDU system. It houses the cables for the control and powering of various electrical and instrumentation equipment. Structural steel framing is provided in the annulus to support and separate cable trays and conduits. In addition, it provides walkways to permit controlled personnel access to the containment, penetrations building, and the reactor service and fuel-handling building.

1.2.6 TURBINE BUILDING

This is a non-seismic Category building constructed of structural steel framing with siding. Safety-related systems and equipment previously contained in this structure were relocated into seismic Category I structures.

1.2.7 SECURITY SYSTEM

The major features of the security system in the HWR design are as follows:

1. A guardhouse is located at the entrance to property perimeter.
2. The entire property and zones within the property boundaries are fenced.
3. All employees and visitors entering and leaving the power plant must pass through a security building. This building contains the monitors and alarms of the security systems.
4. Covered access walkways are provided from the security building to either the CAD building or the turbine building.
5. Access from the CAD building to other structures containing safety-related equipment is possible only from the operating-floor access area in the CAD building.

1.2.8 ENGINEERED SAFETY FEATURES

The systems described below were incorporated into or replaced comparable systems in the CANDU system.

1.2.8.1 Containment Spray System

This system replaces the dousing-tank containment-spray system used in the CANDU design. The spray water is chemically treated to aid in the removal of iodine from the containment atmosphere.

1.2.8.2 High-Pressure Safety Injection (HPSI) System

High-pressure safety injection pumps (HPSIP) and safety injection tanks have been incorporated to provide core recovery for a full spectrum of reactor-coolant-system break sizes, as is the case for the CANDU-600 PHW power plants of recent design. To insure that all reactor-coolant tubes will be cooled during shutdown cooling and recovered after an accident, eight discharge lines from each safeguard pump are included.

The shutdown cooling system will not interface with the containment-spray system. The low-pressure safety injection pumps and shutdown cooling heat exchangers have been designed to meet the CANDU rapid-cooldown requirement. Both the safety injection system and the shutdown cooling system have two separate and redundant trains.

1.2.8.3 Containment Isolation System

Every pipe or duct penetrating the containment is provided with valves to isolate the containment in the unlikely event of a pipe rupture. The isolation-valve operators are designed to fail in a "safe" position should their power source fail.

1.2.8.4 Emergency Water Storage

The condensate-storage tank and the demineralized-water-storage tank are designed to meet seismic Category I requirements. These reinforced-concrete tanks are lined with steel and are buried. The tanks are provided with reinforced-concrete covers designed for missile protection.

1.2.8.5 Ultimate Heat Sink

Redundant cooling towers together with necessary pumps, piping, valves, and heat exchangers are provided. The towers, structures, equipment, and components are designed to meet seismic Category I requirements to insure that they will be available in the unlikely event of an abnormal plant operating condition. The electrical systems will be supplied from redundant sources.

1.2.8.6 Holding Pond

A holding pond is provided for collecting and monitoring all liquid effluents from the plant before they are discharged to the environment.

1.2.8.7 Primary-Component Cooling-Water System

This system is a redundant safety-related system designed to meet seismic Category I requirements. It is a closed system that operates under normal or abnormal plant operating conditions. The system is designed to provide cooling water to the heat exchangers of systems required to perform a safety function or for systems containing tritiated water.

The heat exchangers of the primary-component cooling-water system are connected to the circulating-water cooling towers during normal operation and to the ultimate-heat-sink cooling towers during abnormal plant operating conditions.

1.2.8.8 Combustible-Gas-Control System

A redundant combustible-gas-control system is provided to prevent the accumulation of hydrogen from reaching combustible concentrations in the containment after a loss-of-coolant accident (LOCA).

1.2.8.9 Moderator-Cooling System

An ammonia absorption system is incorporated into the design of the HWR plant. An engineering evaluation of the problems associated with maintaining the moderator pressure and temperature at an acceptable level without pressurizing the calandria led to the selection of a refrigeration system for cooling the moderator fluid.

Since the HWR uses cooling towers and an intermediate closed-circuit cooling loop (PCCW), the economics of cooling the moderator fluid with conventional water-cooled heat exchangers, as used in the CANDU system, became apparent. The cooling system has a duty of 748.2×10^6 Btu/hr and cools 27,100 gpm of heavy water from 160 to 110°F. A conventional mechanical compression system would require large amounts of electricity for the compressor drive motors as well as major maintenance.

The system finally selected is an ammonia absorption system. It reduces the electricity consumption from more than 60,000 to 500 hp. The driving force for refrigeration is furnished by steam from the auxiliary steam boiler.

The problem of ammonia entering the moderator is avoided by the use of double-tube moderator heat exchangers. The moderator is contained in the tube side of the exchanger; the ammonia is in the shell side of the exchanger.

1.2.8.10 Reactor-Coolant System

Protective functions are provided by standard PWR four-channel actuation of Combustion Engineering design with two trips out of four required for the actuation of a protective function.

Stainless-steel wetted surfaces have been used in place of the carbon-steel wetted surfaces of the CANDU design. The advantage of stainless steel is its resistance to corrosion and erosion. The feeder tubes from the headers to the pressure channels will be stainless steel; the major pipe sections, headers, pump casings, and steam-generator primary head and nozzles will be clad on the interior with austenitic stainless steel. Any connecting piping will be austenitic stainless steel.

In accordance with NRC General Design Criterion 4, a complete system of supports for the reactor-coolant system has been incorporated. Supports are included for the steam generators, pumps, major pipe sections, and headers. The supports will be designed to withstand the effects of a LOCA event with a simultaneous safe-shutdown earthquake.

1.2.8.11 Reactor-Coolant Pressure and Inventory Control System

The following systems either were incorporated or replaced comparable systems in the CANDU design:

1. Inventory control, or charging and letdown, was changed to include positive-displacement pumps as presently used in PWR light-water NSSSs. The advantage of this type of pump is a constant flow rate at varying discharge pressures.
2. Overpressure protection was modified to meet the requirements of NRC Regulatory Guide 1.67. This includes power-operated relief valves, which are located upstream (on the reactor-coolant system side) of the surge-line

isolation valves so that they cannot be isolated from the reactor-coolant system. The actuation signal will be provided by protective channels on the reactor inlet header.

3. The pressurizer will be protected from overpressurization when isolated by the addition of a safety valve on the steam space. This valve will be sized to protect the pressurizer and surge line from overpressurization as a result of inadvertent heater actuation when the pressurizer is isolated.
4. So that the surge-line isolation valves meet the single-failure criterion, redundant isolation valves were included in each line to the reactor-coolant-system loops. These valves will be actuated in such a manner that no single failure will prevent at least one valve from closing in the affected loop after a LOCA event.

The system has been modified to include a pressurizer spray for normal operations as well as for cooldown. The spray will inject subcooled heavy water into the steam space at a rate sufficient to prevent a high-pressure reactor trip during any normal plant evolution.

1.2.9 SUMMARY

The modifications indicated in the preceding discussion represent the significant changes that have been incorporated into the conceptual HWR design as compared with the CANDU 600 PHW. Modifications that resulted primarily from increased plant size (1,260 versus 600 MWe) or from the arrangement of the primary-coolant system have not been discussed since they do not represent deviations from the basic CANDU concept. Other modifications to improve plant performance or reduce capital cost, such as slightly enriched fueling or the increase of primary system pressure are discussed in Section 2.4

REFERENCES FOR CHAPTER 1

1. N. Shapiro, J. F. Jesick, et al., Combustion Engineering Study on Heavy Water Reactors (to be published).

2. Atomic Energy of Canada Limited, CANDU 600 Station Design, May 1976.

3. "Point Lepreau," Nuclear Engineering International, June 1977.

Chapter 2

ONCE-THROUGH FUEL CYCLE WITH 1.2% ENRICHED URANIUM-235 AND A BURNUP OF 20,000 MWd/MT

2.1 FUEL-CYCLE DESCRIPTION

This reactor/fuel-cycle combination is a CANDU-type heavy-water reactor (HWR) using 1.2% slightly enriched uranium oxide pellet fuel operating on a once-through cycle. Spent fuel will be stored at the reactor site or at an away-from-reactor storage facility. Ultimately, the spent fuel will be sent to a geologic spent-fuel repository. Low-level waste from fabrication will be sent to a shallow land disposal site.

Pertinent information on the fuel and fuel cycle for the HWR is contained in Tables 2-1 through 2-6.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in the mass-flow diagram (Figure 2-1) and are discussed in the following sections of Volume VII:

Enrichment	Chapter 3
Fuel fabrication 1	Section 4.1
Spent fuel storage	Section 6.3
Waste disposal 1	Section 7.1
Waste disposal 3	Section 7.3

Table 2-1. Fuel-management information

Average capacity factor, %	75
Approximate fraction of core replaced per day, %	0.092
Lag time assumed between fuel discharge and recycle reload, years	2
Fissile material reprocessing loss fraction, %	1
Fissile material fabrication loss fraction, %	1
Yellowcake requirements, MTU/GWe	
Initial core	261.8
Annual equilibrium reload	88.1
30-year cumulative	2775
Separative-work requirements, 10^3 SWU/GWe	
Initial core	0
Equilibrium reload	31.2
30-year cumulative	922.9
Other data for proliferation-resistance assessment	
Fuel-element weight, kg	23
Fresh- and discharge-fuel radiation level, R/hr at 1 meter	
Fresh	0.0003
Discharge (air)	303
Discharge (water)	3.0
Discharge-fuel energy generation rate after 90-day cooling, watts per element	479

Table 2-2. Fuel-assembly volume fractions

Component	Volume fraction
Fuel	0.5114
Coolant	0.4082
Structure	0.0803
Control	--
Total	1.000

Table 2-3. Core-region volume fractions

Component	Volume fraction
Fuel	0.0526
Coolant	0.0420
Structure	0.8128
Control[a]	0.0202
Total	1.000

[a] Control at middle of equilibrium cycle.

Table 2-4. Fuel inventory at the beginning and end of equilibrium cycle

Isotope	Inventory
Uranium-235	652.5
Uranium-236	207.3
Uranium-238	162,669
Plutonium-238	0.6
Plutonium-239	441.5
Plutonium-240	185.7
Plutonium-241	40.4
Plutonium-242	11.5
Fission products and other isotopes	1,844
Americium-241	Not available
Curium-242	Not available
Neptunium-237	406

Table 2-5. Reactor charge data[a]

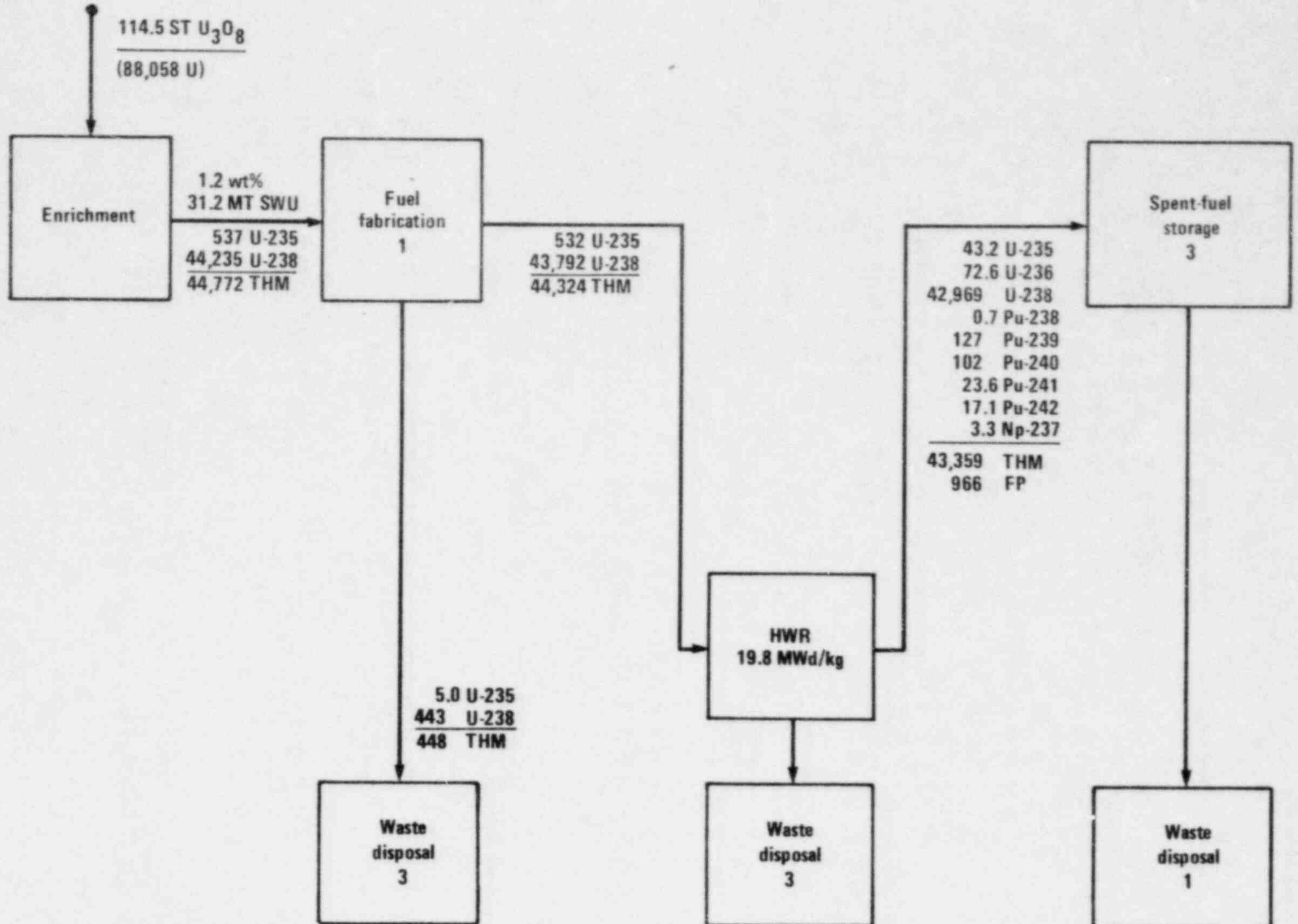
Year	Reactor charge (kg HM)		
	U-235	U-238	Total
1	1,181	164,875	166,056
2	363	29,871	30,234
3	670	55,178	55,848
4	670	55,178	55,848
5	670	55,178	55,848
6	670	55,178	55,848
7	670	55,178	55,848
8	670	55,178	55,848
9	670	55,178	55,848
10	670	55,178	55,848
11	670	55,178	55,848
12	670	55,178	55,[illegible]
13	670	55,178	55,848
14	670	55,178	55,848
15	670	55,178	55,848
16	670	55,178	55,848
17	670	55,178	55,848
18	670	55,178	55,848
19	670	55,178	55,848
20	670	55,178	55,848
21	670	55,178	55,848
22	670	55,178	55,848
23	670	55,178	55,848
24	670	55,178	55,848
25	670	55,178	55,848
26	670	55,178	55,848
27	670	55,178	55,848
28	670	55,178	55,848
29	670	55,178	55,848
30	670	55,178	55,848

[a]For a 1,260-MWe reactor, 75% capacity factor.

Table 2-6. Reactor discharge data[a]

Year	Reactor discharge (kg HM)										Fission products
	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total	
1	79.4	20.7	29,856	0.02	72.2	26.4	4.9	1.12	0.33	30,063	140
2	89.3	66.6	54,585	0.48	150.0	93.3	20.5	12.1	2.5	55,019	828
3	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
4	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
5	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
6	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
7	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
8	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
9	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
10	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
11	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
12	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
13	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
14	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
15	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
16	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
17	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
18	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
19	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
20	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
21	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
22	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
23	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
24	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
25	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
26	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
27	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
28	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
29	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217
30	54.5	91.5	54,141	0.91	160.0	128.5	29.7	21.5	4.2	54,630	1,217

[a]For a 1,260-MWe reactor, 75% capacity factor.



Mass flow in kg per 0.75 GWe-yr.
Abbreviations: FP, fission products; THM, total heavy metal.
Data normalized from a 1,260-MWe reactor.
Equilibrium charge/discharge data from year 22.

Figure 2-1. HWR material-flow diagram for CANDU-type once-through fuel cycle (LEU(5)-OT).

2.2 SAFETY CONSIDERATIONS

2.2.1 GENERAL

For pressure-tube HWRs, as for light-water reactors (LWRs), protection of public health and safety is provided by preventing the dispersion of radioactive material into the environment by means of three barriers: the fuel cladding, the primary heat-transport system (PHTS), and the containment system. The principal purpose of the safety evaluation performed for both LWRs and HWRs is, on the one hand, to determine the degree of integrity maintained by the three barriers under certain postulated safety-related event sequences and, on the other hand, to calculate the potential radiation doses to individuals and the population-at-large from the failure of one or more barriers.

In the design of both LWRs and HWRs, emphasis is placed on the defense-in-depth concept, which includes the following:

1. Designing for safety in normal operation and maximizing the ability to tolerate malfunctions through intrinsic features of sound conservative design, construction, selection of materials, quality assurance, testing, and operation. Margins are incorporated into the plant by adhering to regulatory requirements and the many accepted codes and standards of organizations such as the American Nuclear Society, the American Society of Mechanical Engineers, the American Society for Testing and Materials, and the Institute of Electrical and Electronics Engineers.
2. Anticipating that some abnormal incidents will occur during plant life and making provisions to terminate such incidents and to limit their consequences to acceptable limits, even though important components or systems may fail. Even under these conditions significant margins are provided as a result of using conservative design practices and accepted codes and standards.
3. Providing protection against extremely unlikely events, which are not expected to occur during the life of a single plant, and assuming additional failures of consequence-limiting equipment. From an analysis of these postulated events, features and equipment are designed into the plant to control the postulated events and to insure that there is no undue risk to the public.

Since most nuclear power stations licensed for operation or construction in the United States and elsewhere are of the LWR type, it is logical and helpful to compare, where possible, the safety-related characteristics of HWRs and LWRs. Furthermore, since the conceptual HWR is a modified CANDU-PHW reactor, it is of interest to compare some of the current U.S. Nuclear Regulatory Commission (NRC) licensing requirements for LWRs with those in force for CANDU-PHW reactors in Canada (Refs. 1-31). Section 2.4 of this report assesses the level of safety of HWRs against that of current LWRs and in so doing evaluates the licensing issues that may arise for HWRs in the United States of the design described in Chapter 1.

2.2.2 INTRINSIC SAFETY-RELATED CHARACTERISTICS OF THE HWR

In evaluating the safety of any nuclear power reactor system, it is useful and desirable to distinguish design characteristics that are intrinsic to the nuclear steam supply system (NSSS) from those that are nonintrinsic (i.e., pertaining to subsystems of a more peripheral nature). Among the principal intrinsic safety-related characteristics of HWRs that are substantially different from those of LWRs are the following:

1. The presence of a large heat sink (the moderator) that is intimately dispersed in the core region and is provided with a continuously operating heat-removal system. This latter system will serve as a backup emergency core-cooling system (ECCS) for the highly unlikely event that a postulated low-probability loss-of-coolant accident (LOCA) is followed by failure of the primary ECCS or for the unlikely event of a loss-of-primary-heat-sink accident.
2. Subdivision of the core into a large number of relatively independent power channels. This characteristic allows the primary heat-transfer system (PHTS) to be subdivided into two or more hydraulically independent loops; it also permits the rapid detection and location of any fuel failures.
3. The use of an on-line refueling system, which allows the excess reactivity to be kept low at any time during the fuel cycle and furthermore permits easy on-line replacement of failed fuel.
4. The use of a core lattice with power channels placed at a relatively large pitch, leaving ample space for the introduction of neutron poison devices for control and safety purposes in the low-pressure low-temperature moderator region of the core.
5. The use of a coolant with a positive void coefficient of reactivity (negative coolant-density coefficient of reactivity); loss of all coolant from the core region results in a reactivity insertion of about 1% (at nominal power).
6. An overall power coefficient of reactivity (at nominal power close to zero) that may be either slightly negative or slightly positive.
7. Presence of instability for xenon oscillations, if no external control is applied.
8. A mean neutron lifetime of about 10^{-3} second (i.e., approximately 30 times longer than that for LWRs).
9. The use of a nonferrous material (zirconium-niobium alloy), exposed to the full neutron flux, as part of the pressure boundary of the primary heat-transport system.
10. The use of rolled joints to connect zirconium-niobium alloy pressure tubes to the stainless steel end fittings.

Of the above-listed principal intrinsic safety-related characteristics, item 1 is of particular interest in that one of the main conclusions reached in the Reactor Safety Study (Ref. 32) is the fact that any substantial release of radioactive material to the environment requires prior melting of a substantial part of the core. Since the fuel of the HWR is fully imbedded in a large heat sink, the melting of a substantial core fraction is very improbable.

The possibility of subdividing the primary heat-transport system into two or more hydraulically independent loops (item 2) permits a reduction of the consequences of a postulated low-probability loss-of-coolant accident since only the directly affected loop would blow down its coolant. Complete loss of coolant from a single loop results in a reactivity insertion of less than \$1.00 for a plant with two or more loops.

The capability of rapidly detecting and locating failed fuel (item 2) and easy on-line fuel replacement (item 3) permits the primary heat-transport system to be operated at a very low level of radioactive contamination.

The fact that the excess reactivity is low in an HWR (item 3) limits the amount of reactivity available in a postulated low-probability reactivity accident (e.g., loss-of-regulation accident).

The relatively large space between the power channels in the moderator region (item 4) permits the installation in HWRs of two independent shutdown systems (SDS-1 and SDS-2) with fully separate neutron poison devices; these shutdown systems are located completely outside the pressure boundary of the primary heat-transport system. Moreover, neutron poison devices used for control purposes are kept separately from those serving a direct safety function, resulting in a large degree of separation between control and protection functions.

In considering the potential risk from reactivity accidents, it is of the utmost importance to consider both the maximum amount of reactivity that could be involved and the probability for insertion of this reactivity. The HWR has a positive void coefficient of reactivity for the coolant (item 5); however, the insertion of any appreciable amount of reactivity associated with this coefficient has a very low probability since it requires an initiating event of very low probability (loss-of-coolant accident). Moreover, the total amount is limited (~$1.00), and the effect is reduced by the long mean neutron lifetime (item 8). Moreover, as mentioned earlier, HWRs are equipped with two independent shutdown systems, which would quickly compensate for the positive reactivity introduced, bringing the reactor to a subcritical state. The LWR, on the other hand, has the potential for insertion of larger amounts of reactivity ($1.00) associated with the occurrence of initiating events with a higher probability. As an example, it may be mentioned in this connection that the total reactivity worth of the voids in the core of a boiling-water reactor (BWR) at nominal power is about $7.00, part of which would be introduced on turbine trip.

The characteristic of requiring continuous external automatic control (items 6 and 7) is not limited to HWRs; it is also found in, for example, BWRs, where continuous automatic pressure control is needed to stabilize the system in view of its positive coolant-pressure-reactivity coefficient. A further consideration is that the automatic control required for the HWR can be relatively slow acting since the time constants governing the processes involved (xenon buildup, thermal time constant of the fuel) are relatively large. It is important to note that both the HWR and the BWR are adequately protected against any failure of the automatic control systems.

The use of nonferrous materials in the pressure boundary of the primary heat-transport system of pressure-tube reactors (item 9) is dictated by considerations of neutron economy. The use of rolled joints (item 10) follows from the necessity of connecting two different materials--zirconium-niobium alloy and stainless steel. The LWR, which has been developed in the United States and elsewhere to the level of large-scale commercial deployment, is of the pressure-vessel reactor type and hence does not require the use of nonferrous materials in the pressure boundary of the primary heat-transport system. The development of the nuclear sections of the American Society of Mechanical Engineers (ASME) Pressure Vessel and Boiler Code kept pace with the development of the LWR but did not follow developments in pressure-tube reactors, since there was, up to now, no great need for it. There should, however, be no basic problems with the use of a zirconium-niobium alloy for the pressure tubes in the HWR or with the use of rolled joints. A substantial data base has been developed in Canada and elsewhere on the use of a zirconium-niobium alloy and rolled joints in the primary heat-transport system, including data on the physical properties of a zirconium-niobium alloy at relatively high levels of neutron irradiation. Furthermore, in-service inspection techniques developed for the CANDU reactor enable on-load surveillance, by means of the fueling machines, of the pressure tubes and rolled joints for developing cracks (if any).

The fact that the pressure tubes are exposed to the full neutron flux, so that some degree of embrittlement may be expected with time, requires 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 the LWR, 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 a sudden large-size break in an HWR pressure tube 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, and hence leakage will precede tube rupture ("leak before break") (Refs. 15-18).
2. A leak in 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 pickup system installed 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 by ultrasonic probes used in conjunction with the on-load fueling machine, thus providing an overview of the state of the pressure tubes.
4. Although the pressure tubes are designed to serve for the entire lifetime of the plant, they can be replaced with relative ease, thus permitting early elimination of tubes showing any signs of fault (Ref. 19).

2.2.3 PRINCIPAL ASPECTS OF THE CANADIAN SAFETY APPROACH

The purpose of this section is to summarize the Canadian approach to safety so as to provide an understanding of the design basis for the CANDU-PHW reactor concept. It should be noted, however, that the actual HWR design being presented in this report for NRC review has been modified as discussed in Section 1.2 to improve fuel utilization, to reduce capital cost, and in consideration of U.S. licensing requirements and construction practices.

This section presents an outline of some of the principal aspects of the Canadian safety approach developed for the CANDU-PHW reactor as general background information. The Canadian approach has, from its inception, displayed a tendency toward probabilistic risk assessment. The basic idea is that accidents with a low probability should be allowed to carry larger consequences than accidents with a higher probability. In order to formalize this approach, all systems pertaining to a CANDU-PHW reactor are subdivided into two classes: process systems and safety systems. The first class consists of all systems necessary for the normal operation of the plant: the primary heat-transport system, the reactor-control systems, electrical systems, turbine, etc. The second class consists of all safety systems: the reactor-shutdown systems (SDS-1 and SDS-2), the emergency core-cooling system (often referred to in Canada as the emergency coolant-injection system), and the containment system. It is a design requirement 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--that is, caused by the failure of any one of the process systems--or whether they are of the dual-failure type--that is, caused by the failure of any one of the process systems combined with a simultaneous and independent failure of any one of the safety systems. It should be emphasized here that, except for the containment system, the 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 has established allowable irradiation doses for individuals and for the total population for the two accident and dual-failure accidents, and for the particular plant at the particular site, that the calculated irradiation doses do not exceed the allowable values. Table 2-7 gives the limit doses for single individuals and for the population-at-large for both single- and dual-failure accidents. It also gives Canadian criteria for the maximum frequencies allowable for accidents in the single- and dual-failure categories. The designer is required to demonstrate that the frequency of serious faults in the entire process system is less than 1 3 years and that the unavailability (unreliability) of each safety system is less than 1 in 1,000 years. Safety systems are required to be testable during plant operation. A serious fault in the process system is defined as one that would, in the absence of safety systems, result in a substantial release of radioactive material 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 to the whole body and 3 rem to the thyroid from iodine-131 for individuals and 10^4 man-rem to the whole body and 10^4 man-rem to the thyroid from iodine-131 for the entire population. For accidents in the dual-failure category the maximum allowable frequency of occurrence is 1 in 3,000 years and the allowable reference doses are 25 rem to the whole body and 250 rem to the thyroid for individuals and 10^6 man-rem to the whole body and 10^6 man-rem to the thyroid for the entire population (see Table 2-7). To permit comparison, dose limits in force in the United States are summarized in Table 2-8. It is noted that the allowable Canadian reference doses for accidents in the single-failure category (which includes loss-of-coolant accidents of the maximum size) are smaller by a factor of 50 and 100, respectively, for whole-body and thyroid exposure than those allowable in the United States for a similar accident under 10 CFR 100. On the other hand, however, it should be mentioned that a more conservative radiological source term is used for dose calculations in the United States than in Canada.

Consideration of postulated accidents in the dual-failure category (as defined in Canada) is not a requirement in the NRC licensing procedure. However, such dual-failure accidents are evaluated in the United States on a probabilistic basis (see, for example, Ref. 32).

Some of the principal Canadian safety criteria can be summarized as follows:

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 item 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 one-tenth of the allowable dose to atomic energy workers and the total dose to the population shall not exceed 10^4 man-rem/yr.
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 the exposure of the population at risk shall not exceed 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 25 rem to the whole body, 250 rem to the thyroid, and 10^6 man-rem to the population.
8. In computing doses in items 6 and 7, the following assumptions shall be made unless otherwise agreed to:
   a. Meteorological dispersion that is equivalent to Pasquill category F as modified by Bryant (Ref. 33).
   b. Conversion factors as given by Beattie (Ref. 34).

To provide protection against events that could induce common-mode failures (e.g., fires, airplane crashes, and natural phenomena), CANDU-PHW systems have been subdivided into two groups, powered from physically separate power sources and provided with separate cooling-water supplies:

Group 1	Group 2
Control systems	Shutdown system 2
Safety shutdown system 1	Containment system
Emergency core-cooling system	Emergency power supply
process systems (except auxiliary moderator cooling system)	Emergency water supply
	Emergency instrumentation for plant-status monitoring

Each group of systems is to have the following capabilities: (a) to shut the reactor down to cold conditions, (b) to remove decay heat, and (c) to provide the operating staff with state-of-reactor information. Figure 2-2 shows 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).

It is Canadian practice to consider in the design of CANDU-PHW reactors two levels of severity for seismic events: the design-basis earthquake (DBE) and the site design earthquake (SDE).

The 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." It is applied to nuclear power plant structures that 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 0.2g. 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 exceeding 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 insure that there is no failure of structures, systems, or equipment that are essential to nuclear safety after a seismic event.

The 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 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 that are to be seismically qualified to that level of design earthquake. The minimum ground-motion acceleration for the SDE is 0.03g but it 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 United States. The SDE is comparable in level (not in application) to the operating-basis earthquake (OBE) in the United States.

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

As shown in Figure 2-2, the entire primary heat-transport system of the CANDU reactor is qualified by design for the DBE; this includes the core and pressure tubes, which can withstand earthquakes with a ground acceleration of 0.5g and higher (Ref. 20). For that reason, the Canadian licensing criteria do not require consideration of an earthquake-induced large break in the primary heat-transport system; leaks resulting from an earthquake are, however, accommodated by the design. The Canadian licensing approach for CANDU reactors differs in this area from that in the United States for LWRs, where licensing criteria do require consideration of a large-scale loss-of-coolant accident simultaneously with a seismic event of the severity of the safe-shutdown earthquake. It should be noted, however, that the Canadian requirements concerning protection against earthquakes are in complete agreement with the Codes of Practice and Safety Guides for Nuclear Power Plants of the International Atomic Energy Agency, which are 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 after such an event. While the U.S. safety approach is more conservative than the Canadian with respect to the assumption of the simultaneous occurrence of a maximum-size loss-of-coolant accident 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 the consideration of a large-scale loss-of-coolant occurring simultaneously with an earthquake, the emergency coolant-injection system is not required to be qualified for the DBE. The emergency coolant-injection system is, however, qualified for the SDE, so as to be able to continue to provide core-cooling capability if, during the recovery period after a postulated nonmechanistic large-scale loss-of-coolant accident, an earthquake of the severity level of the SDE (which, by definition, has a relatively "high" frequency of 10^{-2} per year) were to occur.

In view of the above considerations, no credit is taken in the accident analysis for correct functioning of the emergency coolant-injection system after a DBE. In the event of a DBE-induced postulated leak in the primary heat-transport system, core cooling is to be provided by the emergency water supply (EWS), powered by the emergency power supply (EPS); both of these systems are fully qualified by design for the DBE (see Figure 2-2).

It should be noted that the emergency core-cooling system for the HWR conceptual design considered in this report is designed to fully meet U.S. seismic design criteria.

The Canadian licensing criteria require the containment system to be qualified by design for the DBE and other natural phenomena, such as tornadoes and hurricanes. Since Canadian licensing criteria for the CANDU reactor do not require the consideration of a large-scale loss-of-coolant accident following a DBE but only a leak in the primary heat-transfer system, the containment system is designed for DBE-induced loads combined with a coincident containment pressure up to that at the onset of the containment energy removal system-dousing system or vacuum building (approximately 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 United States for LWRs with respect to redundancy, diversity, separation, and independence. This holds particularly true for active components. An exception exists for some passive components (e.g., low-pressure piping), where in some cases Canadian design criteria do not require redundancy for CANDU reactors; this is particularly the case for safety systems with diverse backup systems.

An important characteristic of the Canadian design approach for high reliability is that in many cases "redundancy in safety systems" is provided for protection against certain accident sequences, whereas the approach in the United States is often to provide a single redundant system.

The following are examples of this difference in design approach: 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 2-2) 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 the loss of a coolable-core configuration and core meltdown after a large-scale loss-of-coolant accident (emergency coolant-injection system and moderator cooling system) for CANDU reactors versus one cooling system (emergency core-cooling system) for LWRs.

Table 2-9 shows a nonexhaustive matrix of design-basis accidents of the single- and dual-failure type considered in the Canadian licensing process for Canadian pressurized-heavy-water (deuterium/uranium) (CANDU-PHW) reactors. In the dual-failure category, each type of process-system failure is combined with the failure of any one of the safety systems. These combinations result in trivial cases in some instances, 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 is true particularly for loss-of-coolant accidents combined with containment impairment or for loss-of-coolant accidents combined with failure of the emergency coolant-injection system.

As mentioned in the foregoing, Canadian licensing regulations for CANDU reactors are in some areas more conservative than those in force in the United States 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 U.S. licensing process for LWRs does not require consideration of containment impairment or failure of the emergency core-cooling system in conjunction with a loss-of-coolant accident.

Canadian licensing criteria for CANDU reactors allow the successful operation performance of the emergency coolant-injection system to be assumed for the analysis of a postulated loss-of-coolant accident 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 loss-of-coolant accident combined with failure of the emergency coolant-injection system, the successful operation of the containment system may be assumed. In this case, a considerable fraction of the radioactive fission products may be released from the primary heat-transport system and must be retained by the containment system. It should be mentioned in this connection that failure of the emergency coolant-injection system 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 about 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 United States for LWRs and covers the full spectrum of failures of the primary heat-transport system, up to and including the so-called 100% break of the largest diameter piping. Failures of sufficient magnitude and duration in the primary heat-transport system will result in blowdown of part of (Pickering nuclear station) or the entire (Bruce nuclear station) primary coolant system, depending on whether the primary heat-transport system is divided into two or more independent systems or whether it is a single system. Such loss-of-coolant accidents would require successful operation of the emergency coolant-injection system 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 U.S. analyses of loss-of-coolant accidents 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 offset as for the break postulated in the United States. The outcome of the analysis of loss-of-coolant accidents postulated 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.

The Canadian licensing criteria imposed for CANDU reactors in the evaluation of postulated loss-of-coolant accidents are slightly different from those currently in force in the United States for LWRs. The current U.S. interim ECCS acceptance criteria, having as principal objectives the maintenance of a coolable-core configuration and keeping the energy released by the metal-water reaction at a negligible level, are as follows:

1. The maximum fuel-cladding temperature shall not exceed 1,200°C (2200°F).
2. The calculated oxidation of the cladding shall nowhere exceed 17% of the total cladding thickness before oxidation.

The temperature limit of 1,200°C is imposed on the basis of oxygen embrittlement; melting of the cladding 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 (Refs. 35-37) is that an absolute temperature limit (1,200°C) as part of the oxygen embrittlement criteria (instead of a temperature-time relationship) is extremely conservative. The current Canadian design criteria for the performance of the emergency coolant-injection system therefore do not include a strict limitation to less than 1,200°C but do require that the oxygen concentration be less than 0.7% over at least half the cladding thickness. Figure 2-3 gives an example of the time-temperature relationship for oxygen embrittlement of Zircaloy, showing the difference between Canadian and U.S. criteria. It seems that this Canadian criterion, though perhaps somewhat less conservative than its U.S. counterpart, does meet the intent of the U.S. ECCS acceptance criteria--namely, the avoidance of excessive embrittlement of the Zircaloy cladding.

The U.S. regulatory guidelines with respect to the release of radioactive material from the fuel and the containment after a loss-of-coolant accident are as follows (Ref. 38):

1. Twenty-five percent 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 iodines.
2. One hundred percent 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 instantaneous release of 25% of all radioactive iodine and 100% of all radioactive noble gas is an overly conservative assumption since (apart from the prompt release of a portion of the free fission products in the fuel-rod gaps) the release of fission products from the fuel matrix is governed mainly by the fuel temperatures attained and by the fuel-temperature distributions in the fuel rods and in the core as a whole. For the various 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 the release of radioactive material from the fuel rods after a loss-of-coolant accident therefore requires

1. Calculation of cladding and fuel-temperature transients after coolant blowdown and initiation of cooling by the emergency coolant-injection system.
2. Determination of the failure fraction of fuel-rod cladding (criteria for cladding failure: (a) 5% uniform strain or (b) excessive oxygen embrittlement because of oxygen concentrations greater than 0.7% over at least half the cladding thickness).
3. Calculation of the quantity of fission products released from fuel rods with failed claddings on the basis of the calculated spatial temperature transients in the fuel attained during blowdown and subsequent cooling by the emergency coolant-injection system. It is conservatively assumed that the reactor has been operated continuously at 100% power before the loss-of-coolant accident.

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

2.2.4 FREQUENCY CLASSIFICATION OF SAFETY-RELATED EVENTS[a]

The range of safety-related events considered can be subdivided into three groups described as follows:

A. Events of moderate frequency (anticipated operational occurrences) leading to no abnormal radioactivity releases from the facility.
B. Events of small probability with the potential for small radioactivity releases from the facility.
C. Potentially severe accidents of extremely low probability, postulated to establish the performance requirements of engineered safety features and used in evaluating the acceptability of the facility site.

It is highly desirable, for both safety and economic reasons, that group A (moderate-frequency) events, such as partial loss of forced reactor-coolant flow, should result in reactor shutdown with no radioactivity release from the fuel and with the plant capable of readily returning to power after corrective action. Analysis and evaluation of these moderate-frequency conditions offer the opportunity of detecting and correcting faults in a particular plant design that might otherwise lead to more serious failures. Safety is certainly enhanced if all those events that can be identified as having a reasonable chance of occurring are shown to be covered by features designed to prevent their occurrence and significant damage.

The second group of events, such as a complete loss of forced reactor-coolant flow or partial loss of reactor coolant from small breaks or cracks in pipes, must be shown to present minimal radiological consequences. The actual occurrence of such accidents may, however, prevent the resumption of plant operation for a considerable time because of the potential for failure of the cladding of some fuel rods and the consequent requirement for replacement and cleanup.

Evaluation of these postulated safety-related events must show that under accident conditions the engineered safety features and containment barriers function effectively to eliminate (or reduce to an insignificant level) the potential for radioactivity releases to the environment. In this way, assurance is gained that these unlikely events would lead to little or no risk to public health and safety. These studies also show the effectiveness of safety features designed into the facility to cope with unlikely accidents and show the margins of safety that exist in the design by indicating the types of failure that can be accommodated.

To provide additional safety margins, extremely unlikely accidents of group C are postulated in spite of their low probability and the steps taken to prevent them. The hypothetical accidents evaluated during the safety review of power reactors include loss of reactor coolant resulting from postulated major ruptures in the primary-coolant-system piping (loss-of-coolant accidents).

In these types of accidents, the potential exists for breaching of the fuel-rod cladding and the release of radioactive material from the reactor fuel, transport of a portion of this radioactive material through leakage paths in the containment barriers, and, finally, leakage of some portion of it to the environment. Each type of accident is analyzed to determine whether there is assurance that adequate safety features

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[a]See also Reference 39.

have been engineered into the plant, in the form of passive barriers or active systems, to limit the consequences of a release of fission products from the reactor fuel and to show that the maximum radiological doses would not exceed the values specified in 10 CFR 100, even under highly pessimistic assumptions.

2.2.5 ANALYSIS PARAMETERS

Work performed to date for this Preliminary Safety and Environmental Information Document (PSEID) does not cover detailed analyses of postulated safety-related events.

2.2.6 TRIP SETTINGS

Work performed to date for this PSEID does not cover the detailed analysis necessary for a selection of reactor trip settings.

2.2.7 RADIOLOGICAL PARAMETERS

Work performed to date for this PSEID does not cover an evaluation involving radiological parameters.

2.2.8 COMPUTER PROGRAMS

The present PSEID does not cover detailed analysis of postulated safety-related events.

2.2.9 GROUP A EVENTS

Work performed to date for this PSEID does not cover detailed analysis of postulated safety-related events.

2.2.10 GROUP B EVENTS

Work performed to date for this PSEID does not cover detailed analysis of postulated safety-related events.

2.2.11 GROUP C EVENTS

Work performed to date for this PSEID does not cover detailed analysis of postulated safety-related events.

Table 2-7. Reference radiological dose limits in Canada

Plant condition	Maximum frequency allowed	Climatic conditions 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 to whole body; 3 rem/yr to thyroid[a]	10^4 man-rem/yr to whole body; 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 stability category if local data incomplete	0.5 rem to whole body; 3 rem to thyroid	10^4 man-rem to whole body; 10^4 man-rem to thyroid
Process equipment failure plus failure of any safety system	1 per 3,000 years	Either worst weather existing at most 10% of time or Pasquill F stability category if local data incomplete	25 rem to whole body; 250 rem to thyroid[b]	10^6 man-rem to whole body; 10^6 man-rem to thyroid

[a]For other organs use one-tenth of the occupational doses recommended by the International Commission on Radiation Protection (ICRP).

[b]For other organs use five times the ICRP annual occupational dose (tentatively).

Table 2-8. Radiological dose limits in the United States

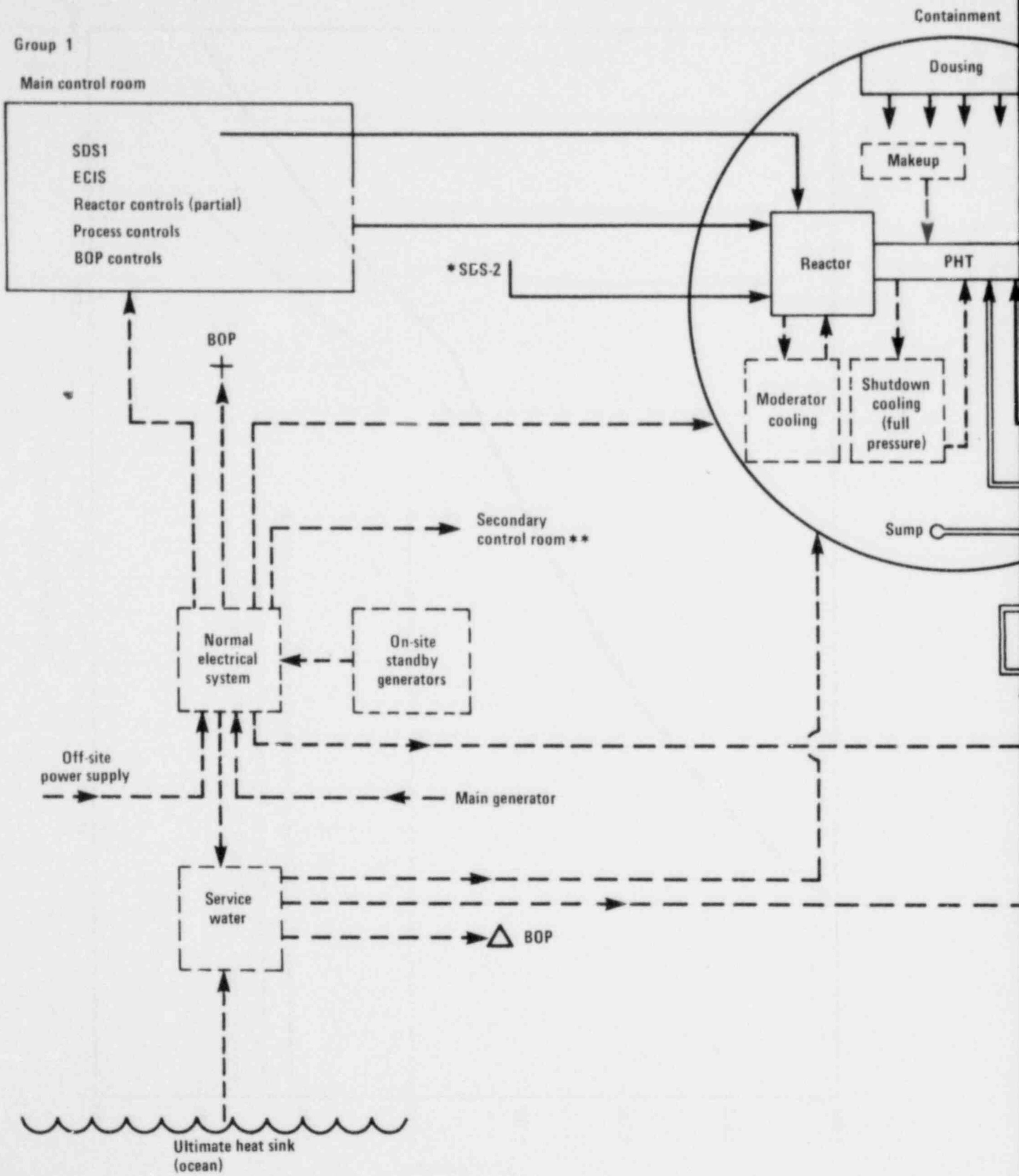
U.S. Nuclear Regulatory Commission regulations	U.S. Environmental Protection Agency regulations
Permissible levels of radiation in unrestricted areas (10 CFR 20.105) For average radiation levels and anticipated occupancy: 0.5 rem/yr to whole body Radiation level causing dose of 2 mrem in 1 hour or 100 mrem in 7 consecutive days As low as reasonably achievable from effluent releases, 5-mrem/yr target Reactor site criteria for major accidents (10 CFR 100.11) Dose at site boundary in first 2 hours: 25 rem to whole body, 300 rem to thyroid Dose in low-population zone during cloud passage: 25 rem to whole body, 300 rem to thyroid	Planned discharges (normal operation) 0.025 rem/yr to whole body 0.075 rem/yr to thyroid 0.025 rem/yr to any other organ Per GWe-yr 50,000 Ci Kr-85 5 mCi I-129 0.5 mCi Pu-239 and other alpha-emitting transuranics

Table 2-9. Accident matrix[a]

Process-system failure	Single-failure accidents	Dual-failure accidents			
		SDS-1	SDS-2	ECIS	Containment
Loss of regulation	X	X	X	--	--
Loss of coolant	X	X	X	X	X
Loss of (primary) heat sink[b]	X	X	X	--	--

[a]Postulated accidents indicated by an X require analysis, whereas postulated accidents indicated by a dash are trivial cases.

[b]Postulated loss-of-heat-sink accidents require assessment of alternative heat sinks.


Group 1
Main control room
SDS1
ECIS
Reactor controls (partial)
Process controls
BOP controls
Containment
Dousing
Makeup
Reactor
PHT
*SDS-2
BOP
Moderator cooling
Shutdown cooling (full pressure)
Sump
Secondary control room**
Normal electrical system
On-site standby generators
Off-site power supply
Main generator
Service water
BOP
Ultimate heat sink (ocean)

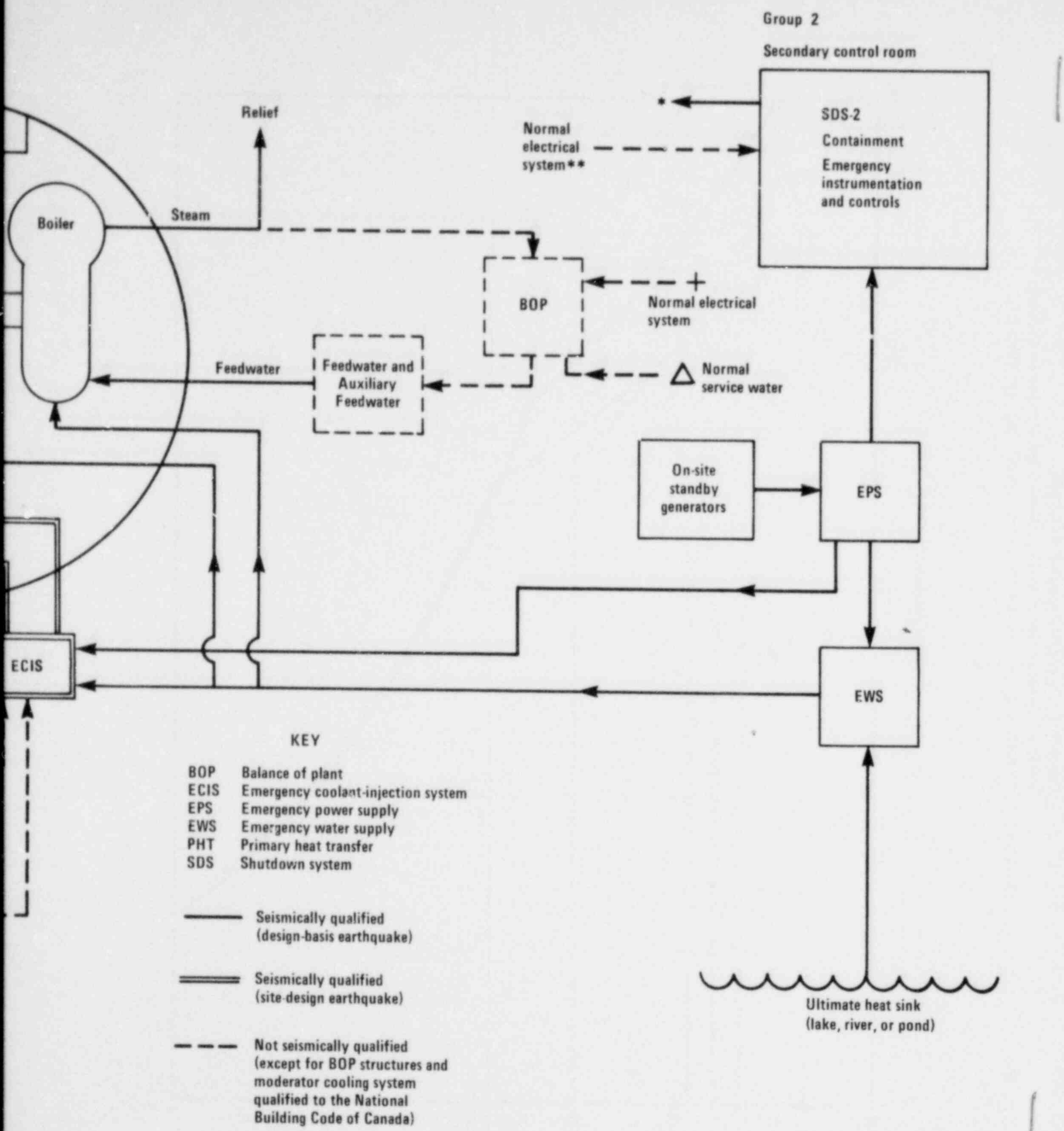


Figure 2-2. Seismic qualifications and system separation for a 600-MWe CANDU-PHW station at an ocean site.

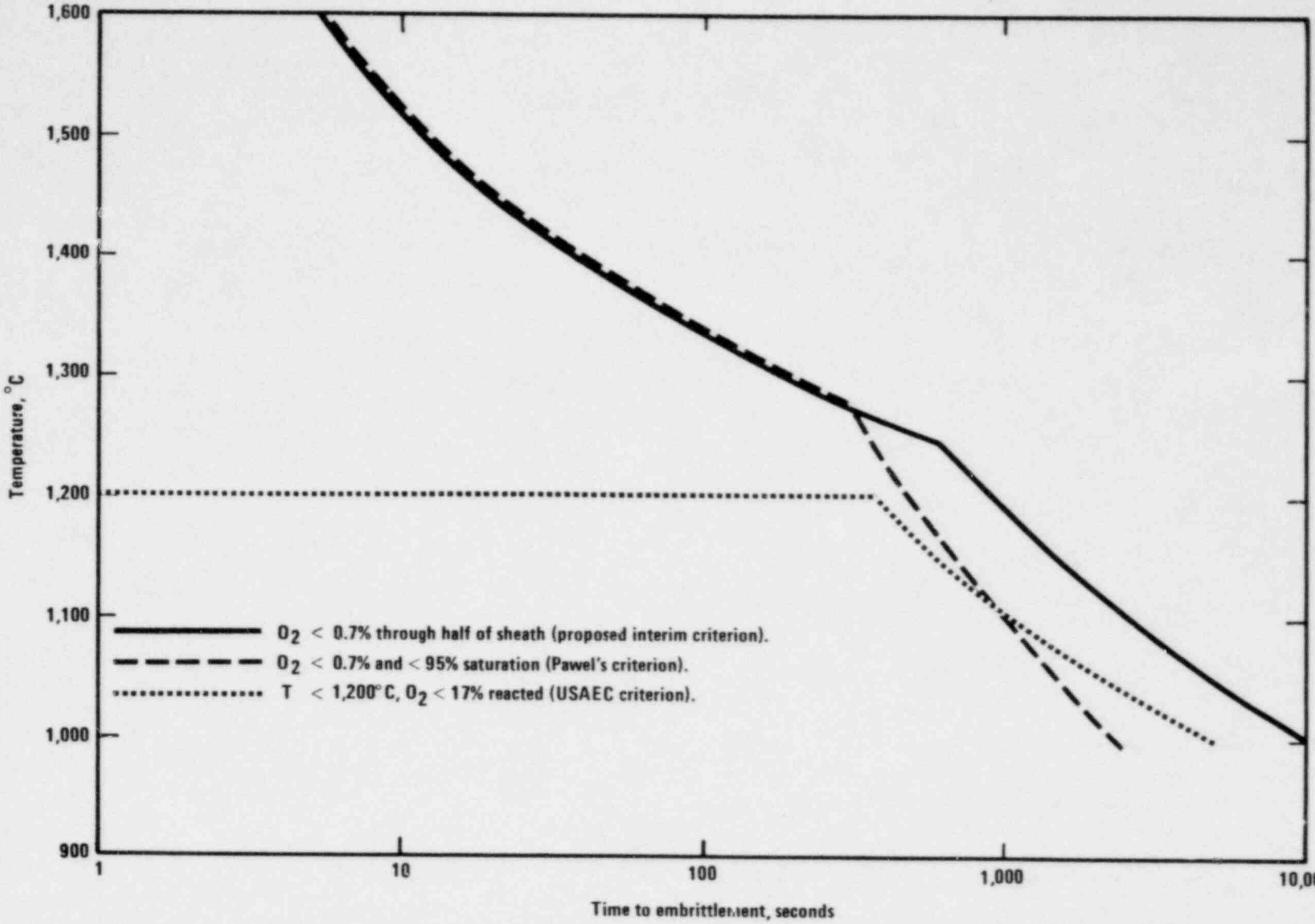


Figure 2-3. Example of the time-temperature relationship for embrittlement as determined by embrittlement criteria for isothermal oxidations of 0.42-mm-thick cladding.

2.3 ENVIRONMENTAL CONSIDERATIONS

2.3.1 SUMMARY ASSESSMENT

The thermal efficiency of the conceptual HWR is less than that of the reference LWR. The impacts of thermal effluents and chemical and biocidal releases (which are related principally to the operation of the heat-dissipation system) are therefore higher than those for the reference LWR. These somewhat larger releases and consequent impacts are, however, not large enough to have a major effect on siting and licensability.

The predicted radioactive releases are smaller than the values for the reference LWR with the exception of tritium. The predicted tritium releases, to the atmosphere and to the hydrosphere, are larger than the corresponding releases from the reference LWR. The net effect on doses from gaseous releases is to make them somewhat smaller than those for the reference LWR. The doses from liquid releases are, however, greater (see Section 2.3.9).

2.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM (RG 4.2/3.2)

The conceptual HWR design differs from the Canadian CANDU-PHW plant primarily by an increase in thermal output to 4,029 MWt, the use of a higher burnup fuel cycle with 1.2-wt% enriched uranium fuel, modifications in coolant conditions to increase plant efficiency, and design changes made to facilitate conformance with U.S. design or licensing practices.

Where possible, experience and features of the CANDU-PHW reactor are retained to take advantage of the extensive development and proved performance of the Canadian HWR design. Basic parameters describing the conceptual HWR plant are given in Table 2-10.

2.3.3 STATION LAND USE

There are no specific features of the HWR plants that would indicate differences in land use from that of LWR plants.

2.3.4 STATION WATER USE (RG 4.2/3.3)

The principal single use of water at the HWR plant is for makeup to the heat-dissipation system. Much smaller amounts are required for plant use (after demineralization) as well as for such uses as laundry, showers, and sanitary facilities (see Table 2-11). As shown in Table 2-12, the maximum and average rates at which makeup is required are about 12,850 and about 7,600 gpm, respectively, at 1,000 MWe. In comparison, the reference LWR requires 11,500 gpm and 6,800 gpm, respectively.

2.3.5 HEAT-DISSIPATION SYSTEM (RG 4.2/3.4)

Any of various heat-dissipation systems may be used for an HWR plant, depending on site conditions and other factors. One of the more commonly used systems is a wet natural-draft cooling tower. Such a system, with freshwater makeup, was assumed for this report.

A typical natural-draft cooling tower for a 1,000-MWe unit has a single shell with a height of about 522 feet and a maximum shell diameter of about 404 feet.

Heat is dissipated to the atmosphere by evaporation and by sensible-heat transfer. Evaporation predominates, but the balance between the two depends on air temperature and humidity. The average rate of water use, therefore, varies from month to month. Blowdown is needed to limit the concentration of solids in the circulating water. For the plant discussed here, a maximum concentration factor of 5 is used, although other values are frequently found. Design data for the heat-dissipation system are shown in Table 2-12 for a site in the north-central United States.

Circulating water will be periodically chlorinated to control algae and other slime-forming microorganisms. Typically, chlorine is added as required to achieve a residual chlorine content of 0.5 to 1.0 ppm for 1 to 2 hours per day. The cooling-tower blowdown may have a small residual chlorine content during periods of chlorination.

2.3.6 RADWASTE SYSTEM AND SOURCE TERMS (RG 4.2/3.5)

Sources of radioactivity, release paths, and radioactive-waste-processing systems are described briefly in this section. The principal assumptions and parameters that were used in calculating the quantities of radioactivity that would be released from an HWR plant are listed in Table 2-13. Because these parameters are for a plant with a capacity of 1,260 MWe, the results were normalized to 1,000 MWe for the reference HWR design.

An HWR sited in the United States will conform to the requirements of Appendix I to 10 CFR 50 to insure that releases of radioactive material to unrestricted areas during normal reactor operations are kept as low as is reasonably achievable. The application of existing PWR technology is assumed to be acceptable.

2.3.6.1 Source Terms (RG 4.2/3.5.1)

The sources of radioactivity in the HWR plant are fission products and materials in the reactor core and coolant that have been activated by neutron irradiation. Small amounts of fission products are released to the reactor coolant through defects in the fuel cladding, and activated core materials are released to the coolant by corrosion.

The capability, in the HWR system, for detection, location, and removal of failed fuel under load, allows operation with low levels of fission products in the primary heat-transfer system.

The performance of CANDU-type HWR fuel is well documented from CANDU reactor operation in Canada and elsewhere over the last decade. However, the extension of CANDU-type fuel operation from discharge burnups of 7,500 MWd/MTU to approximately 20,000 MWd/MTU is not well known even though current CANDU-type fuel has been irradiated up to burnups of 15,000 MWd/MTU with satisfactory results (Ref. 40).

The environmental concerns over the use of modified CANDU-type fuel focus on several factors. Among these are the anticipated failure rate, the radioactive materials released to the coolant in the event of a fuel failure, and the portion of such materials that is released to the environment.

Because of the higher fuel burnups, the average in-core inventory of long-lived radioactive fission products for the HWR fuel would increase to about 300% of that available with current CANDU-type fuel operation. However, this increased fission-product concentration is still less than that found in LWRs (with discharge burnups near 30,000 MWd/MTU). The consequences of the higher fission-product concentration will depend on the security of the barriers between the fuel and the environment and between the fuel and the public. For safety reasons, the HWR must provide the same protection as LWRs, and the release of gaseous and liquid radioactive materials will be less than that of LWRs.

At present, the anticipated fuel-failure rate can only be estimated from Canadian public data sources (Ref. 40). The early CANDU-fuel failure rate from all causes was reported as 27 in 10,000 bundles (there are 28 to 37 fuel rods per bundle, depending on plant design), and the primary fuel-failure mechanism was attributed to "power ramping" (or pellet-cladding interactions). Power ramping of the fuel is an inherent operational maneuver of the CANDU on-line refueling scheme. After the adoption of operating procedures specifically aimed at mitigating the pellet-cladding interaction, the fuel-failure rate was reduced to less than 10 in 10,000 bundles. More recently the Canadians have introduced CANLUB, a graphite-based lubricant on the fuel pellet surface as a means of further reducing the rates of fuel failure related to pellet-cladding interactions (Ref. 41).

The fact that CANDU-type fuel is subject to fuel failures related to pellet-cladding interactions at burnups below 7,500 MWd/MTU may be consequential for HWR fuel operation at higher burnups. This is especially so because the dominant mechanism is considered to be stress-corrosion cracking of the Zircaloy cladding, induced by the release of such fission products as iodine from the fuel.

Sources of radioactivity were assumed to be such as to give the same concentrations of fission-product isotopes (except for tritium) in the HWR coolant as in the LWR coolant with 0.25% failed fuel. In view of the failed-fuel removal capability of the HWR system, this assumption is considered to be very conservative.

Tritium is of particular interest in the heavy-water reactor. It is present from ternary fissions but in the HWR comes chiefly from neutron reactions with deuterium in the reactor coolant and moderator by the D(n,l)T reaction. The resultant tritium-production rate in the moderator and coolant is about 2,800 Ci/MWe-yr.

Figure 2-4 shows the accumulated tritium activity in the coolant and moderator over the 30-year operation of the plant. The activities have been calculated from the following tritium production equations (Ref. 42):

Moderator

$$^{3}\text{H atoms/barn-cm} = 2.06 \times 10^{-6} \left(1 - e^{-0.05622t}\right)$$

Primary Coolant

$$^{3}\text{H atoms/barn-cm} = (2.21 \times 10^{-10}) + (5.8 \times 10^{-8}) \left(1 - e^{-0.05622t}\right)$$

There is no chemical cleanup for tritium; however, fission-product radioactivity is removed from the reactor coolant by cleanup in the makeup and purification system and by small amounts of leakage. Figure 2-5 shows the potential paths for the transfer of radioactivity to other plant systems.

For the calculation of tritium release from the plant it is assumed that the leakage of D_2O is 500 g/hr, of which 450 g/hr originates from the primary coolant and 50 g/hr from the moderator. In addition, it is assumed that two-thirds of the tritium source is in the gaseous release and one-third in the liquid release. The fission-product source term calculation is the same as that used for the reference LWR, with the exception that the leakage rates to the secondary system and to the containment building are 23.8 and 2.6 lb/day, respectively, versus 110 and 24 lb/day for the reference LWR.

Figure 2-6 shows the steam and power-conversion system components that are most important from the standpoint of radioactivity in the system and releases to the environment. Noble gases and small amounts of iodine that leak into the steam generator are carried out with the steam, pass through the turbine and condenser, and are removed from the condenser by the air-removal system. A filter system removes most of the iodine from the gases to be discharged to the atmosphere. Noble gases and iodine also reach the atmosphere directly in a small amount of steam leakage.

Nonvolatile radioactive materials collect in the steam-generator liquid. They are removed in the blowdown stream, which goes to the condenser and there mixes with the condensate. About 65% of the condensate stream passes through the condensate-polishing demineralizer as it is returned to the steam generator. Thus, nonvolatile radioactive isotopes are collected in the condensate-polishing demineralizers.

The cost of heavy-water upkeep as well as a reduction in radioactivity release has motivated high recovery efficiencies (greater than 95%) in rooms that contain heavy-water systems in existing CANDU plants. Upkeep includes replacing lost heavy water ($100/lb) and upgrading what is recovered at less than reactor-grade isotopic purity. Experience in upkeep improvement has led to application of the following design steps to minimize the leakage of heavy water and maximize collection and recycling:

1. Reduction in the number of valves in heavy-water systems
2. More extensive use of bellows-sealed valves
3. Reduction in the number of mechanical joints
4. Improved packing arrangements in valves
5. Reduced use of light water in heavy-water areas
6. Improvement in the efficiencies of driers

Achievable heavy-water total upkeep has been shown to be about 70 lb/day, with a loss of less than 10 lb/day in the CANDU Pickering-type plant.

Figure 2-7 shows the heavy-water leakage recycle system that collects and processes (for recycling) water from the moderator-coolant and the reactor-coolant systems.

2.3.6.2 Liquid-Radwaste System (RG 4.2/3. .)

The miscellaneous-liquid-waste system (Figure 2-8) processes liquid wastes from the sources described above as well as from other sources: laundry and shower wastes, equipment drains, and floor drains. Laundry and shower wastes and condensate from the containment coolers are collected and monitored. If there is no significant radioactivity, these wastes are discharged, with the laundry and shower wastes being filtered before discharge. If significant radioactivity is present, these streams are routed to the equipment discussed below for processing.

Waste to be processed is collected in waste tanks and passed through particulate and carbon filters to remove oil and other organics. It then goes to an evaporative waste concentrator. The concentrates (bottoms) are sent to the solid-waste-handling system for solidification and disposal. The distillate is passed through an ion exchanger and then stored in a waste-condensate tank for monitoring and discharge.

Turbine-building drainage is collected and discharged. Quantities of important isotopes, calculated by the GALE computer code, are shown in Table 2-14. These data are extrapolated from calculations for a PWR of similar size. The assumptions used in these calculations (e.g., flow rates) are shown in the tables.

Discharges from the miscellaneous-liquid-waste system are directed to the river, lake, ocean, or other body of water on which the plant is sited.

2.3.6.3 Gaseous-Waste System (RG 4.2/3.5.3)

The gaseous-waste management system is shown in Figure 2-9. Compressed storage is provided for gases removed from the gas stripper of the heavy-water recycle system, the volume-control tank, and the reactor drain tank. The gas from the first is deuterium containing small (volumetrically) amounts of fission products. The gas from the reactor drain tank is nitrogen cover gas, displaced as the tank is filled. A recombiner is provided to remove deuterium or oxygen from the stored gases. The deuterium is removed in the recombiner; the small volume of fission-product gases that is left is returned to one of the storage tanks for long-term holdup. The gases from the volume-control tank can be processed similarly.

Nitrogen cover gas displaced by filling the reactor drain tank is compressed in the gaseous-waste-management system. It is stored for reuse as a cover gas.

In addition to these major sources of radioactive gases, there are the leakage paths discussed earlier. These are small leaks from the reactor-coolant system to the containment, small leaks of reactor coolant to the auxiliary building, and small leaks from the reactor-coolant system to the steam and power-conversion system.

The containment is equipped with an internal recirculating filter system containing particulate, absolute, and charcoal filters. This system removes particulate and iodine activity before containment purge. The containment is vented or purged through similar filter systems.

The auxiliary-building ventilation system also contains particulate, absolute, and charcoal filters. This system filters air exhausted from areas that might become contaminated by reactor-coolant leakage. Most of the gaseous activity leaking into the steam and power-conversion system is contained in air removed from the condenser. This effluent is also filtered by particulate, absolute, and charcoal filters. Total calculated gaseous releases of radioactivity have been extrapolated, and the results are shown in Table 2-15.

2.3.6.4 Solid-Radwaste System (RG 4.2/3.5.4)

Materials transferred to the solid-radwaste system for disposal include spent demineralizer resins and evaporator concentrates. These are solidified for offsite disposal. Other solid wastes such as contaminated clothing, papers, and filters are also sent away from the site for disposal. It is estimated that a total of two hundred

55-gallon drums will be shipped off the site for disposal each year. In comparison, 1,050 fifty-five-gallon drums of low-level waste are estimated to be shipped off-site for the reference LWR.

2.3.6.5 Comparison of HWR and Reference LWR

Tables 2-16 and 2-17 show the estimated annual releases of liquid and gaseous effluents from the reference HWR and the reference LWR plant. Both plants have been normalized to 1,000 MWe for the comparison.

2.3.7 CHEMICAL AND BIOCIDAL WASTES (RG 4.2/3.6)

The main sources of chemical and biocidal wastes are the cooling-tower blowdown stream and the chemical effluents from the regeneration of demineralizers that treat makeup water. The cooling-tower-blowdown stream contains dissolved solids that enter the makeup stream and are concentrated by evaporation during operation of the cooling towers. This stream will also intermittently contain a small chlorine residual from chlorination of the condenser cooling water (Section 2.3.5).

Acid and caustic soda solutions are used for demineralizer regeneration. These wastes are held up and neutralized before discharge. They contain no radioactivity.

2.3.8 EFFECTS OF OPERATION OF THE HEAT-DISSIPATION SYSTEM (RG 4.2/5.1)

The natural-draft cooling tower is the system assumed for all the Nonproliferation Alternative Systems Assessment Program (NASAP) alternatives; therefore the effects are qualitatively the same for all cases. The systems differ only in the amount of heat released. The HWR has a lower thermal efficiency than the reference LWR; the quantity of heat dissipated is 10% greater than for the reference LWR. The effects would therefore be correspondingly greater than for the reference LWR. The difference of 10% is significant but would not be a major problem for siting and licensing.

2.3.9 RADIOLOGICAL IMPACT OF ROUTINE OPERATION (RG 4.2/5.2)

The contributions to radiation doses from liquid effluents are given in Table 2-18; the contributions from noble-gas releases and from radioiodines and particulates are given in Tables 2-19 and 2-20, respectively. Because of the increasing release of tritium during the life of the plant, doses were calculated at two chronological points of plant operation, the 13th year and the 30th year. The doses are given as ratios to those calculated for the LWR reference plant with the underlying assumption that all site-related factors are the same for both plants as described in Sections 2.3.1 and 2.3.9 of PSEID, Volume I. Under these assumptions the doses from gaseous releases are somewhat smaller than those for the reference LWR with the exception of the dose to "child's thyroid" as indicated in Table 2-20.

The doses from liquid releases are greater by a factor of 8.8 for the child's whole body and by 1.5 for the critical organ at the 13th year of operation. The same factors are 14 and 2.3 respectively for the 30th year of operation.

The liquid-pathway doses are due primarily from tritium as indicated in Table 2-18.

It should be noted that the doses from liquid releases depend to a great extent on the site-specific dilution factor which enters into the calculations. Reported operating experience for CANDU reactors (Ref. 43) indicates that liquid effluents contribute approximately 0.1% of ICRP concentration limits at the Canadian site.

2.3.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES (RG 4.2/5.3)

As discussed in Section 2.3.7, the major chemical and biocidal discharges in the HWR plant are from cooling-tower operation and the regeneration of demineralizers that treat makeup water. These discharges are similar in kind and quantity to those from the reference LWR. The cooling-tower discharges would tend to be somewhat greater than those for the reference LWR because the heat-dissipation system is larger (by 10%). The effects of chemical and biocidal discharges would be larger than those for the reference LWR in proportion to the quantities discharged. These differences are not large, however, and should not result in significant difficulties in siting and licensing.

2.3.11 OCCUPATIONAL EXPOSURE

Compilations and studies of historical data show that workers in PWR plants are exposed to an integrated radiation dose that averages 400 to 500 man-rem/yr-unit. Most of this dose is incurred in maintenance and repair activities; much smaller amounts are incurred in reactor operation, waste processing, and refueling (see Table 2-21). These data also show significant numbers of individuals with exposures in the range of 1 to 10 man-rem.

Exposures in an HWR plant are affected significantly by the degree of leaktightness of the plant because of the high content of tritium in the moderator and the coolant. For the four 500-MWe Pickering A reactors, the occupational exposure from tritium in 1974 was 120 man-rem/yr-unit compared to a total exposure of 410 man-rem/yr-unit (Ref. 44). The average exposure was 1 man-rem/yr-unit. These exposure rates have been decreasing over recent years until in 1978 the values were 100 man-rem/yr-unit from tritium and 205 man-rem/yr-unit for total exposure (Ref. 45). With a design aiming at minimizing coolant losses for economic reasons and special personnel protection, the annual exposures in a HWR are not expected to be very different from those experienced in LWRs.

Table 2-10. Basic parameters for the conceptual HWR plant

Fuel cycle	Once-through
Burnup, MWd/MTU	20,000
Base reactor thermal output, MWt	4,029
Electrical output, MWe	1,260
Normalized output, MWe	1,000
Heat rate, Btu/kW-hr	10,913
Heat-dissipation rate (at normalized output), Btu/hr	7.5×10^9

Table 2-11. Estimated water use in the conceptual HWR plant

Flow	Volume (gpm)
Makeup to cooling-tower system (maximum)	16,000
Makeup to cooling-tower system (average)	9,500
Input to laundry, hot showers, sanitary and potable water	3
Input to demineralized-water system	140
Demineralized-water-system waste	10

Table 2-12. Heat-dissipation-system design data for a wet natural-draft cooling tower

Heat-dissipation rate (maximum, full power), Btu/kW-hr	7.5×10^9
Evaporation and drift (maximum, full power), gpm	12,850
Evaporation and drift (annual average), gpm	7,600
Blowdown (maximum), gpm	3,350
Blowdown (annual average), gpm	1,900

Table 2-13. Principal parameters and conditions used in determining the releases of radioactive material from a 4,029-MWt HWR plant

Principal plant parameters	
Operating-power fission-product source term, %	0.25
Primary system	
Mass of coolant, lb	8.85×10^5
Mass of moderator, lb	8.70×10^5
Letdown rate, gpm	176
Leakage rate to secondary system, lb/day	23.8
Leakage rate to auxiliary building, lb/day	160
Leakage rate to containment building, lb/day	2.6
Frequency of degassing for cold shutdowns, per year	2
Secondary system	
Steam flow rate, lb/hr	1.76×10^7
Mass of steam per steam generator (4), lb	0.8×10^4
Mass of liquid per steam generator, lb	0.6×10^5
Secondary coolant mass, lb	2.8×10^6
Rate of steam leakage to turbine building, lb/hr	1.7×10^3
Dilution flow, gpm	4.0×10^3
Containment-building volume, ft^3	3.2×10^6
Frequency of containment purges, per year	4
Recirculation system	
Flow rate, cfm	1.8×10^4
Operating period per purge, hr	16
Mixing efficiency, %	73

Decontamination factors for liquids

Nuclide	Boron recycle	MLWMS[a]	SGB/VCC[b]
Iodine	1×10^5	1×10^4	1×10^2
Cesium, rubidium	2×10^4	1×10^5	1×10^1
Molybdenum, technetium	1×10^5	1×10^6	1×10^4
Yttrium	1×10^4	1×10^5	1×10^2
Others	1×10^6	1×10^5	1×10^2

System	All nuclides except iodine	Iodine
Waste evaporator	10^4	10^3
Boron-recovery-system evaporator	10^3	10^2

Table 2-13. Principal parameters and conditions used in determining the releases of radioactive material from a 4,029-MWt HWR plant (continued)

Demineralizers	Cation[c,d]	Anion[c,d]	Cesium, rubidium
Mixed-bed demineralizers			
Li_3BO_3	10	10	2
H^+OH^-	10^2 (10)	10^2 (10)	2 (10)
Cation demineralizer	10^2 (10)	1 (1)	10 (10)
Anion demineralizer	1 (1)	10^2 (10)	1 (1)
Powdex	10 (10)	10 (10)	1 (10)

Removal by plateout

Nuclide	Removal factor
Molybdenum, technetium	10^2
Yttrium	10

Iodine partition factors (gas/liquid)

Leakage to containment building	0.1
Leakage to auxiliary building	0.005
Steam leakage to turbine building	1.0
Steam generator (carryover)	0.01
Main condenser air ejector	0.0005

[a]Miscellaneous-liquid-waste management system.

[b]Steam-generator blowdown/volatile coolant chemistry (condensate treatment).

[c]For two demineralizers in series, the decontamination factor for the second demineralizer is given in parentheses.

[d]Does not include cesium, molybdenum, yttrium, rubidium, and technetium.

Table 2-14. Liquid radioactive source terms normalized for a 1,000-MWe HWR plant[a]

Nuclide	Source term (Ci/yr)	Nuclide	Source term (Ci/yr)
Bromine-82	0.00002	Cesium-138	0.00001
Bromine-83	0.000026	Barium-139	0.0000
Rubidium-86	0.00001	Barium-140	0.00005
Strontium-89	0.00005	Lanthanum-140	0.000005
Strontium-91	0.00001	Cerium-141	0.000005
Yttrium-91m	0.000001	Cerium-143	0.000002
Yttrium-91	0.000026	Praseodymium-143	0.000005
Zirconium-95	0.000005	Cerium-144	0.00001
Niobium-95	0.000005	Praseodymium-144	0.000005
Molybdenum-99	0.00008	Neodymium-147	0.000002
Technetium-99m	0.00008	Sodium-24	0.00003
Ruthenium-103	0.000001	Phosphorus-32	0.000005
Rhodium-103m	0.000001	Phosphorus-33	0.00003
Tellurium-125m	0.000001	Chromium-51	0.0001
Tellurium-127m	0.000025	Manganese-54	0.00001
Tellurium-127	0.00005	Manganese-56	0.000026
Tellurium-129m	0.000133	Iron-55	0.00008
Tellurium-129	0.00008	Iron-59	0.00005
Iodine-130	0.0001	Cobalt-58	0.0008
Tellurium-131m	0.00013	Cobalt-60	0.0001
Tellurium-131	0.000025	Nickel-65	0.000005
Iodine-131	0.0371	Niobium-92	0.00002
Tellurium-132	0.00265	Tin-117m	0.000005
Iodine-132	0.00265	Tungsten-185	0.000005
Iodine-133	0.0265	Tungsten-187	0.00013
Iodine-134	0.00002	Neptunium-239	0.00005
Cesium-134m	0.00001	All others[b]	0.0001
Cesium-134	0.00265		
Iodine-135	0.005	Total[c]	0.08
Cesium-136	0.0013		
Cesium-137	0.00265	Tritium	9,000[d]
Barium-137m	0.00265		14,500[e]

[a]Fuel failure assumed to be 0.25%.

[b]Some nuclides with discharges of less than 10^{-5} Ci/yr-unit are not identified but are included in the "all others" term.

[c]Except tritium.

[d]Leakage at 13-year concentration, 500 g/hr (90% primary coolant, 10% moderator), represents one-third of the total tritium release.

[e]Leakage at 30-year concentreation, 500 g/hr (90% primary coolant, 10% moderator), represents one-third of the total tritium release.

Table 2-15. Gaseous radioactive source terms normalized for a 1,000-MWe PWR plant

Nuclide	Total source term (Ci/yr)
Krypton-83m	<1
Krypton-85m	2.9
Krypton-85	100
Krypton-87	<1
Krypton-88	3.7
Krypton-89	<1
Xenon-131m	11.7
Xenon-133m	21.2
Xenon-133	1,908
Xenon-135m	1
Xenon-135	13.3
Xenon-137	<1
Xenon-139	0.26
Iodine-131	0.0133
Iodine-133	0.016
Tritium	18,000[a]
	29,000[b]
Carbon-14	1.6
Particulates	0.013

[a]Leakage at 13-year concentration, 500 g/hr (90% primary coolant, 10% moderator), represents two-thirds of the total tritium release.

[b]Leakage at 30-year concentration, 500 g/hr (90% primary coolant, 10% moderator), represents two-thirds of the tritium release.

Table 2-16. Liquid radioactive effluents from the HWR plant and the reference LWR plant (normalized to 1,000 MWe)

Nuclide	Radioactivity released (Ci/yr)		Nuclide	Radioactivity released (Ci/yr)	
	HWR	LWR		HWR	LWR
Bromine-82	0.00002	0.00007	Barium-137m	0.00265	0.01
Bromine-83	0.000026	0.0001	Barium-139	0.00001	0.00004
Rubidium-86	0.00001	0.00004	Barium-140	0.00005	0.0002
Strontium-89	0.00005	0.0002	Lanthanum-140	0.000005	0.0001
Strontium-91	0.00001	0.00006	Cerium-141	0.000005	0.00002
Yttrium-91m	0.000001	0.00002	Cerium-143	0.000002	0.00001
Yttrium-91	0.000026	0.0001	Praseodymium-143	0.000005	0.00002
Zirconium-95	0.000005	0.00002	Cerium-144	0.00001	0.00005
Niobium-95	0.000005	0.00002	Praseodymium-144	0.000005	0.00002
Molybdenum-99	0.00008	0.003	Neodymium-147	0.000002	0.00001
Technetium-99m	0.00008	0.003	Sodium-24	0.00003	0.0001
Ruthenium-103	0.000001	0.00002	Phosphorus-32	0.000005	0.00002
Rhenium-103m	0.000001	0.00002	Phosphorus-33	0.00003	0.0001
Tellurium-125m	0.000001	0.00001	Chromium-51	0.0001	0.0003
Tellurium-127m	0.000025	0.0001	Manganese-54	0.00001	0.00006
Tellurium-127	0.00005	0.0002	Manganese-56	0.000026	0.001
Tellurium-129m	0.000133	0.0005	Iron-55	0.00008	0.0003
Tellurium-131m	0.00013	--	Iron-59	0.00005	0.0002
Tellurium-131	0.000025	0.0001	Cobalt-58	0.0008	0.003
Tellurium-132	0.00265	0.01	Cobalt-60	0.0001	0.0004
Iodine-130	0.0001	0.004	Nickel-65	0.000005	0.00002
Iodine-131	0.0371	0.14	Niobium-92	0.00002	0.00006
Iodine-132	0.00265	0.01	Tin-117m	0.000055	0.00002
Iodine-133	0.0265	0.1	Tungsten-185	0.000005	0.00002
Iodine-134	0.00002	0.00007	Tungsten-187	0.00013	0.0005
Iodine-135	0.005	0.02	Neptunium-239	0.0001	0.0002
Cesium-134m	0.00001	0.00003			
Cesium-134	0.00265	0.01	Tritium	9,000[a]	270[b]
Cesium-136	0.0013	0.005		14,500[c]	
Cesium-137	0.00265	0.01			
Cesium-138	0.00001	0.00002			

[a]Thirteenth-year operation.
[b]Annual average.
[c]Thirtieth-year operation.

Table 2-17. Gaseous effluents from HWR plant and the reference LWR plant (normalized to 1,000 MWe)

Nuclide	Radioactivity released (Ci/yr)	
	HWR	LWR
Krypton-23m	< 1	1
Krypton-85m	2.9	11
Krypton-85	100	380
Krypton-87	< 1	2
Krypton-88	3.7	14
Krypton-89	< 1	1
Xenon-131m	11.7	44
Xenon-133m	21.2	80
Xenon-133	1,908	7,200
Xenon-135m	< 1	1
Xenon-135	13.3	50
Xenon-137	< 1	--
Xenon-139	< 1	--
Iodine-131	0.0133	0.05
Iodine-132	--	--
Iodine-133	0.016	0.06
Tritium	18,000[a] 29,000[c]	580[b]
Carbon-14	1.6	6
Particulates	0.013	0.05

[a]Thirteenth-year operation.
[b]Annual average.
[c]Thirtieth-year operation.

Table 2-18. Contributions to radiation doses from liquid effluents

Nuclide	Contribution (%) to organ dose	
	Child's whole body	Critical organ
Tritium	99	83
Iodine-131	--	14
Iodine-133	--	1
Others	1	2
Ratio of dose to that from reference LWR	8.8[a]	1.5[a]

[a]Column A corresponds to values at the 13th year of operation and column B at the 30th year of operation.

Table 2-19. Contribution to doses from various noble gases

Nuclide	Contribution (% to organ dose) Whole body	Skin
Krypton-83m	(a)	(a)
Krypton-85m	(a)	(a)
Krypton-85	(a)	9
Krypton-87	1	1
Krypton-88	8	4
Krypton-89	2	2
Xenon-131m	(a)	(a)
Xenon-133m	1	2
Xenon-133	83	77
Xenon-135m	(a)	(a)
Xenon-135	4	3
Xenon-137	(a)	1
Ratio of dose to that from reference LWR	0.27	0.27

[a]Less than 1%.

Table 2-20. Contribution to doses from various radioiodines and particulates

Nuclide	Contribution (% to organ dose)[a] Infant's thyroid		Child's thyroid	
	A	B	A	B
Iodine-131	55	43.5	20	7.41
Iodine-133	1	0.53	0.3	0.19
Carbon-14	1	0.7	1	0.77
Tritium	43	55.5	79	85.5
Ratio of dose to that from reference LWR	0.47	0.44	1.19	1.29

[a]Column A corresponds to values at the 13th year of operation and column B at the 30th year of operation.

Table 2-21. Distribution of radiation exposure by activity (1975 data)

Activity	Fraction of exposure (%)
Reactor operations	11
Maintenance	72
In-service inspection	3
Waste processing	7
Refueling	8

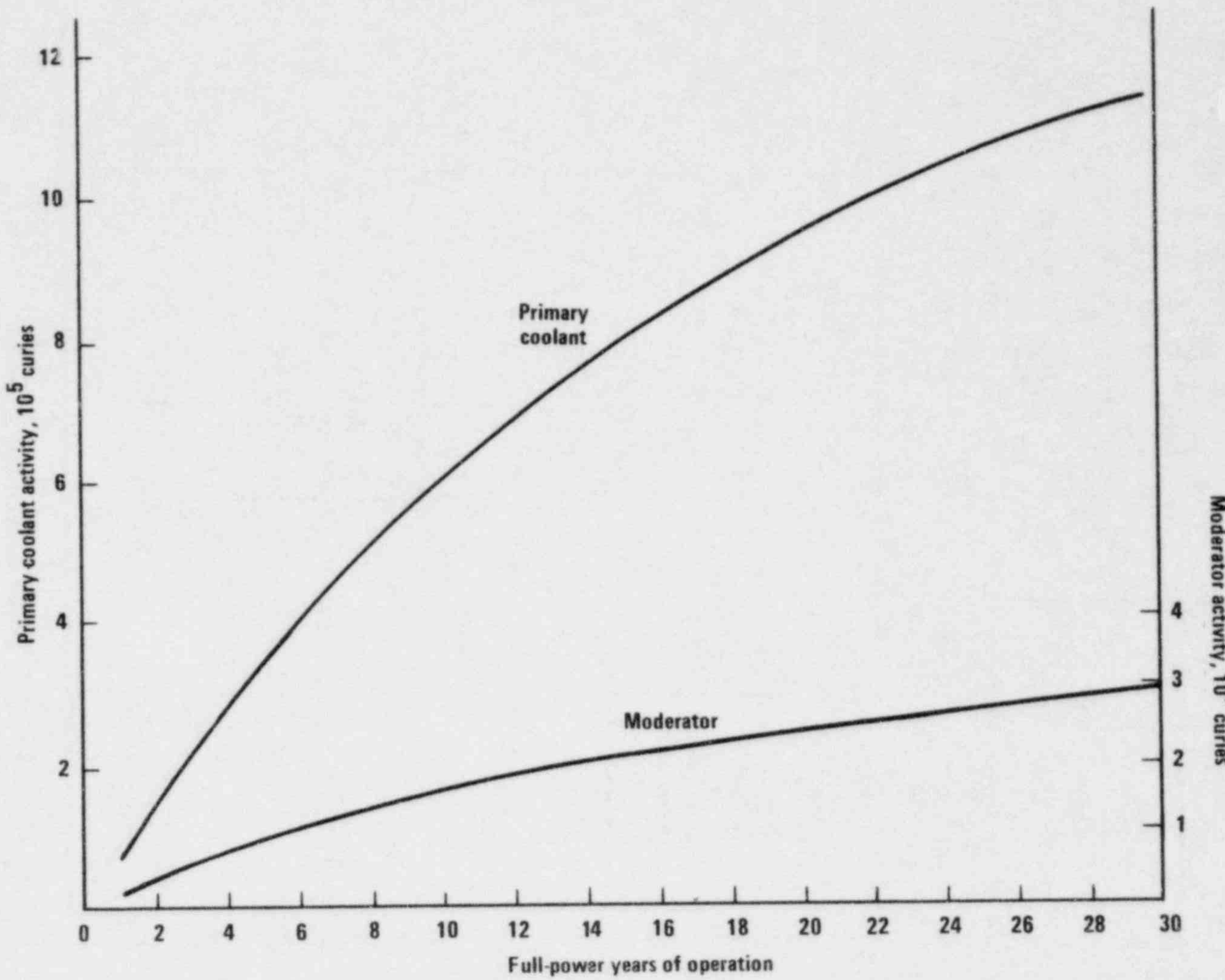


Figure 2-4. Tritium activity in primary coolant and moderator.

1072 ii

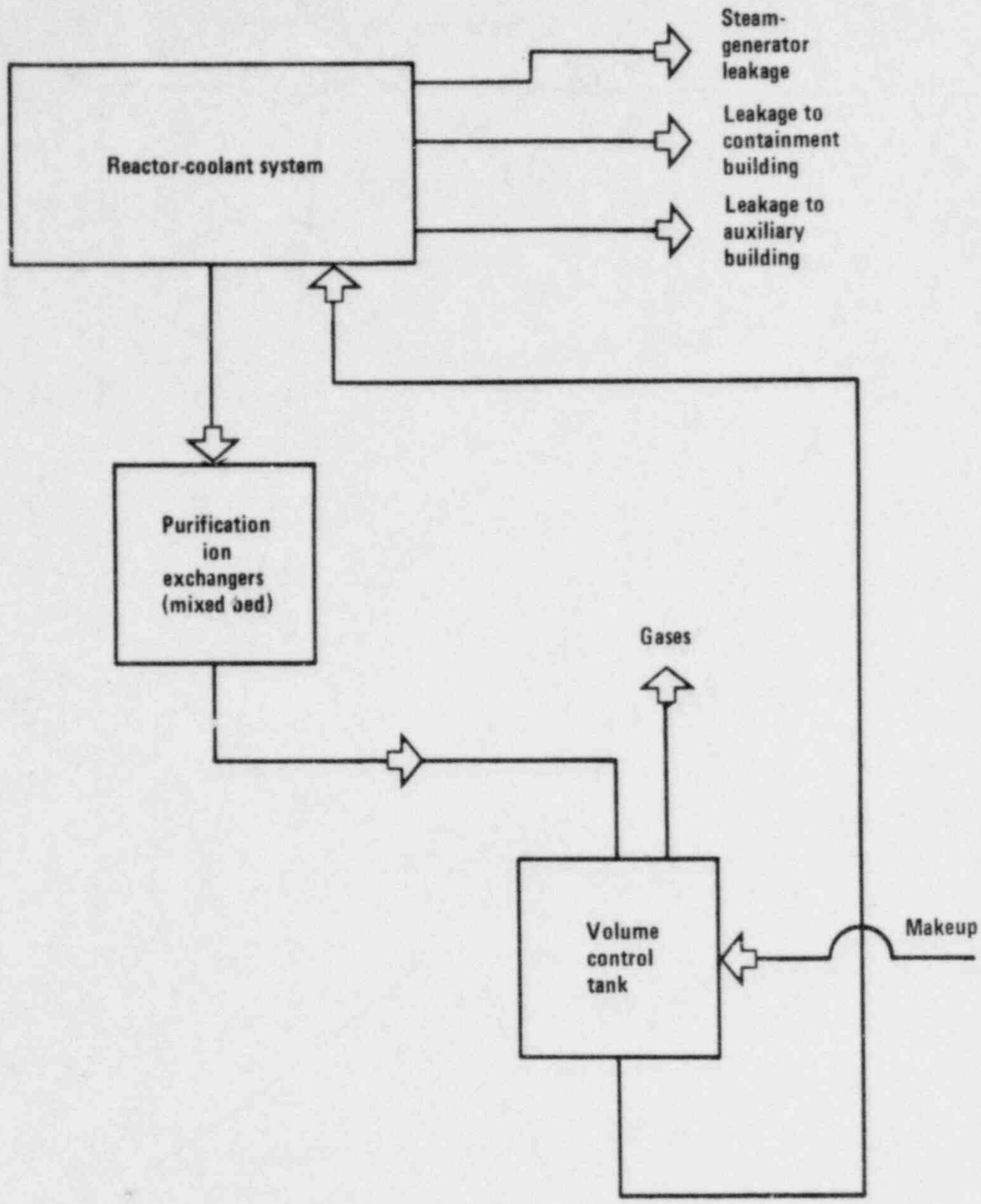
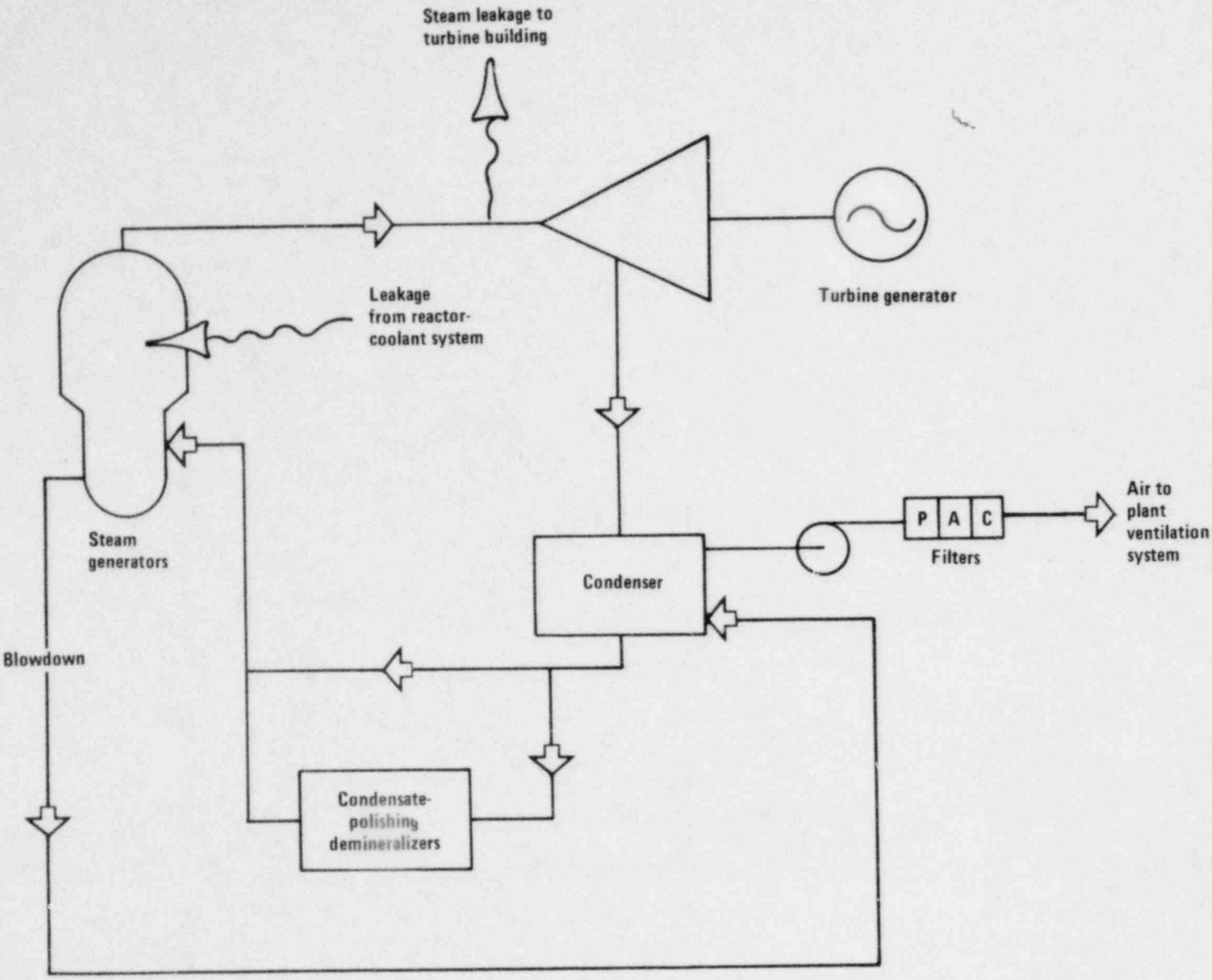


Figure 2-5. Potential paths for the transfer of radioactivity from the reactor-coolant system to other plant systems.

1072 II



Abbreviations: P = particulate; A = air; C = carbon.

Figure 2-6. Steam and power-conversion systems with sources of radioactivity.

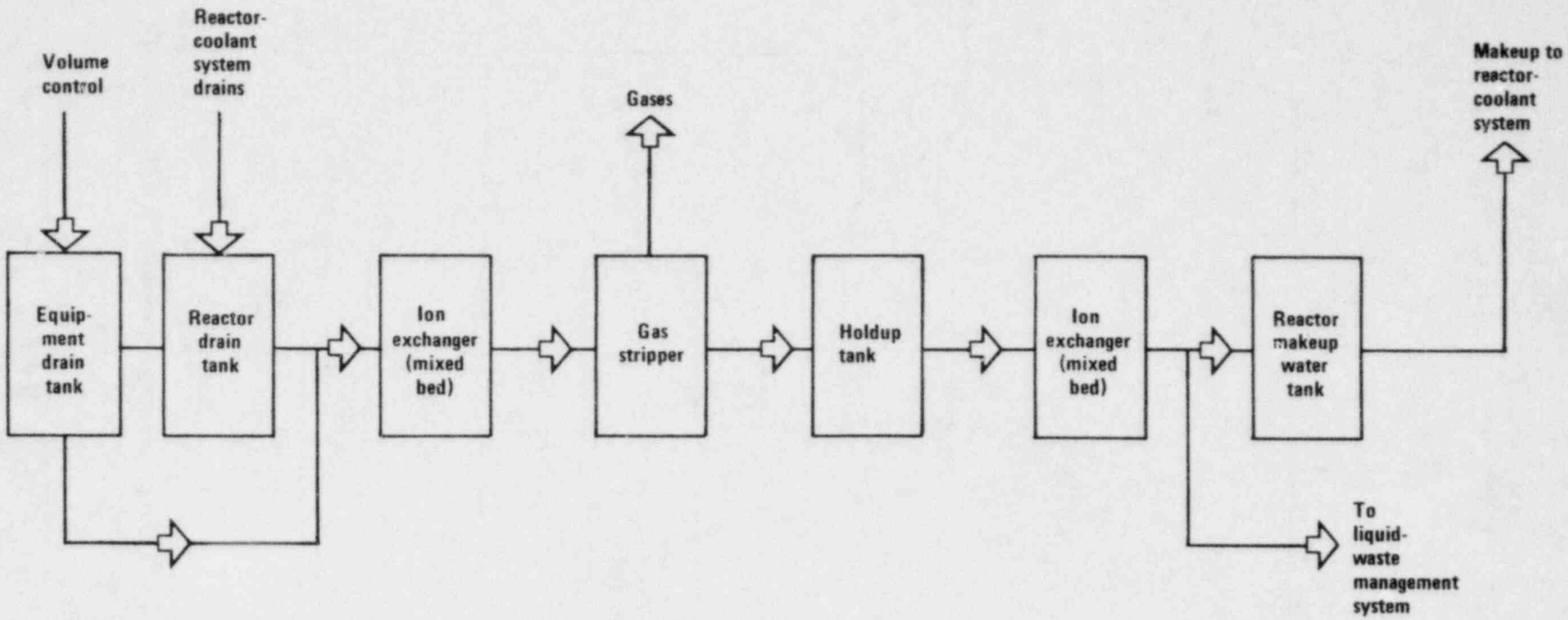


Figure 2-7. Heavy-water leakage recycle systems.

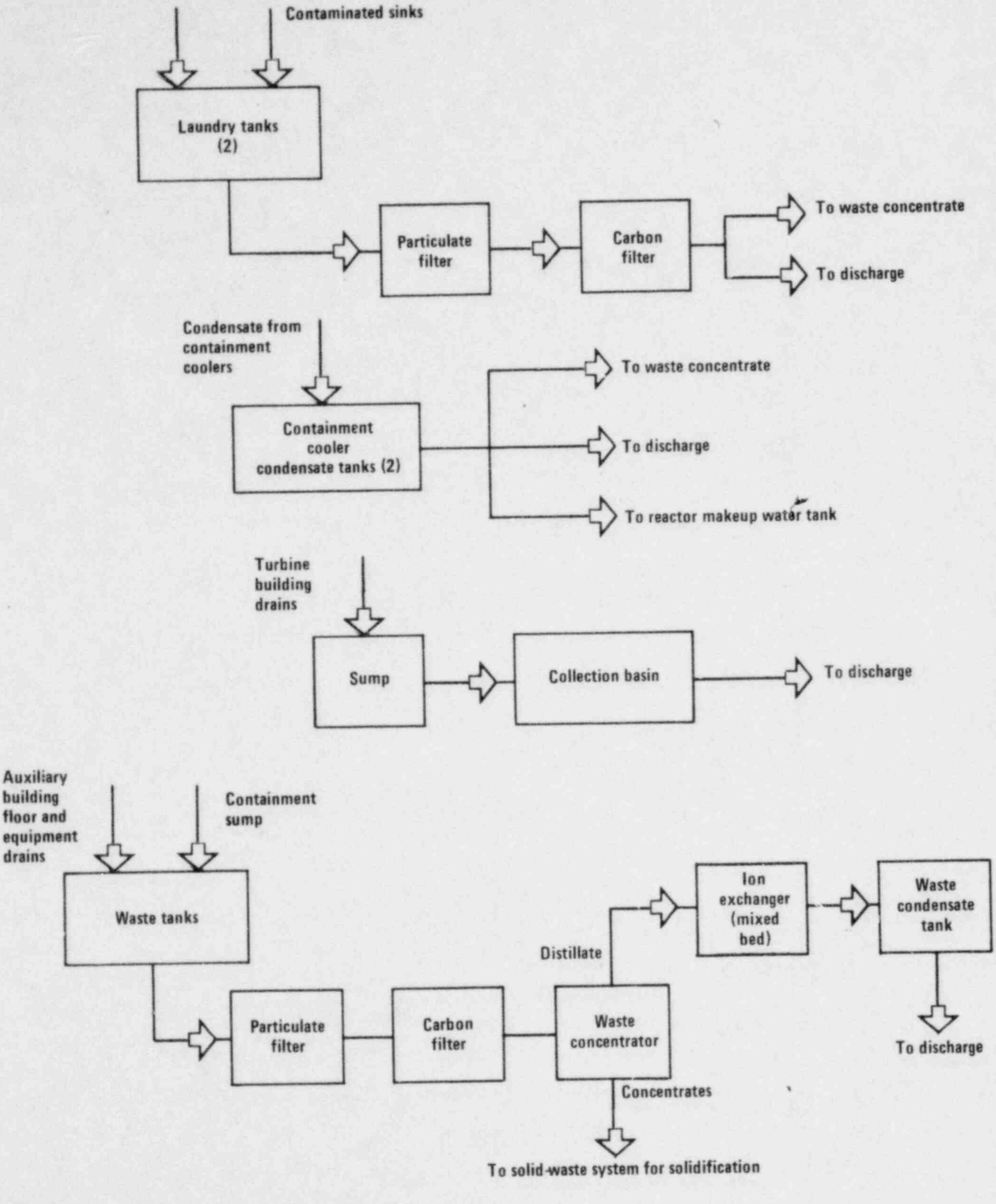
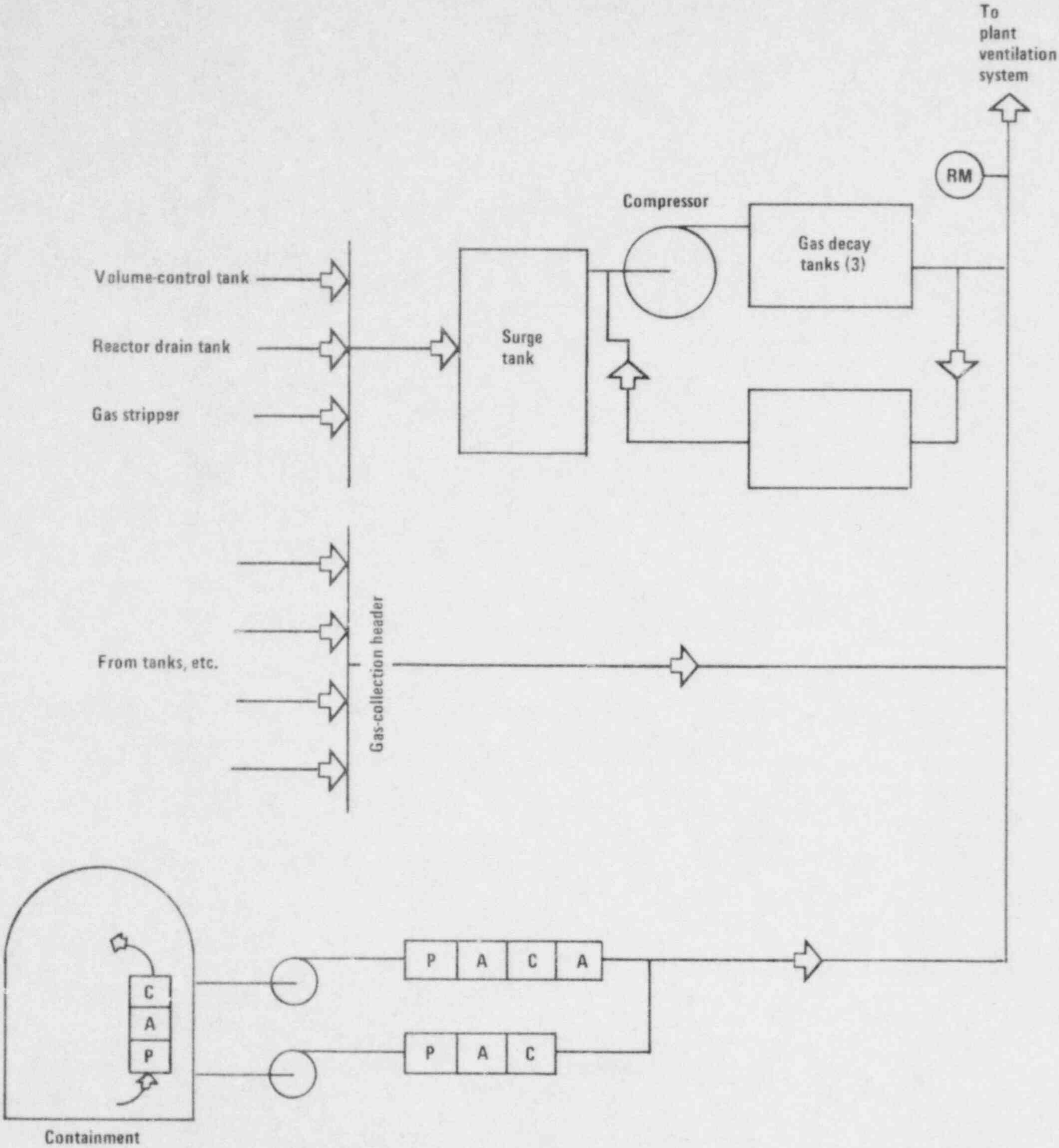


Figure 2-8. Miscellaneous-liquid-waste management.



Abbreviations: RM = radiation monitor; P = particulate; A = air; C = carbon.

Figure 2-9. Gaseous-waste system.

2.4 LICENSING STATUS AND CONSIDERATIONS

2.4.1 GENERAL

No Safety Analysis Report (SAR) for an HWR in the United States has been completed. Although the preparation and NRC review of an SAR for the HWR would be instrumental in establishing the licensability of the concept and in identifying design changes that may be required for siting an HWR in the United States, the effort to prepare an SAR is beyond the scope of the present evaluation and would be considerable. Therefore this section is limited to a qualitative discussion of safety-related aspects with the objectives of (a) establishing the prospects for HWR licensing in the United States and (b) identifying areas in which additional research and development (if any) may be required to resolve potential licensing issues.

The conceptual HWR design described in preceding sections of this chapter is based on the CANDU-PHW reactor. A number of changes, however, have been made to the basic CANDU-PHW design, the most notable of which are the following:

1. The use of slightly enriched fuel
2. An increase in primary and secondary system pressures
3. An increase in net electrical generating capacity

In viewing the safety and licensability of the HWR, it is useful to separate the question of licensability of the CANDU-PHW reactor, as currently deployed in Canada, from the question of the licensability of the various design changes to the CANDU-PHW reactor that have been incorporated into the reference conceptual design of the HWR.

Reactors of the standard CANDU-PHW design are, of course, licensed and operating in Canada as well as in several other nations. The extensive analyses performed for the CANDU-PHW by the designer, Atomic Energy of Canada Limited; the review and acceptance of these analyses by the Canadian licensing authority, the Atomic Energy Control Board; and the highly satisfactory performance of CANDU-PHW reactors testify to the overall safety and licensability of the concept. However, as mentioned earlier, it should be recognized that, to a large degree, Canadian licensing requirements have developed independently of those in the United States for the LWR, and hence some changes in HWR design may be necessary to conform to U.S. licensing traditions and construction practices, even though such changes would not necessarily result in greater safety margins.

As pointed out in Section 2.2, in considering the safety of any reactor type, it is useful to distinguish between intrinsic safety features and the nonintrinsic features of the plant. The intrinsic safety features include such aspects as core transient performance (e.g., coefficients of reactivity) and fundamental design characteristics (e.g., the use of pressure tubes) that are very difficult or costly to alter by design changes. The nonintrinsic features of the plant are more readily altered to meet various licensing and design criteria.

A review of the noninstrinsic features of the CANDU-PHW reactor was performed by United Engineers & Constructors (UE&C) for the U.S. Department of Energy (DOE) (Ref. 46). A number of design changes that were considered necessary for conformance with U.S. licensing criteria and construction practices were identified by UE&C. These engineering changes were subsequently incorporated into a conceptual design for a large HWR plant. These evaluations by UE&C indicated no fundamental problem with modifying the nonintrinsic features of the CANDU-PHW system to conform to appropriate U.S. criteria, although the recommended design changes have a significant impact on plant capital cost.

The reference conceptual HWR design, discussed in detail in Chapter 1, used the UE&C studies as a point of departure. It therefore includes those changes in engineered design features that were found by Combustion Engineering, Inc./United Engineers and Constructors to be necessary for conformance with U.S. licensing criteria and/or design practices.

As pointed out in Section 2.2, current NRC regulatory requirements (design criteria, guidelines, etc.) were to a very large extent developed for the LWR, having in mind their intrinsic safety-related characteristics. If the NRC regulatory requirements had been developed for a reactor type other than the LWR, with different intrinsic safety-related characteristics, then most probably these regulatory requirements would have been worded differently, even though ultimate intent would most probably have been the same. Therefore, it is not at this point clear that the design changes assumed necessary by UE&C would in point of fact be required by NRC for an actual HWR power plant construction license in the United States.

A review of licensing issues associated with a CANDU-PHW plant sited in the United States, giving due attention to the intrinsic safety features of the concept, was performed at the Argonne National Laboratory (Ref. 47). This report concluded that relatively few modifications to the CANDU reactor would be required to obtain conformance with the intent of U.S. licensing criteria and design codes. In this report emphasis was placed on evaluating the CANDU design with respect to the intent of U.S. criteria and design codes, in recognition of the fact that the CANDU reactor is considered to be at least as safe as the LWR and because the regulatory criteria and codes were developed for the LWR and hence are not entirely appropriate to the HWR. The HWR concept was also the subject of a brief review by the NRC (Ref. 48). This review, while identifying a number of areas that may require additional and more detailed evaluation in order for NRC to fully understand the CANDU safety design basis, concluded that it was likely that a suitably designed HWR could be licensed in the United States. Finally, a design study, including a safety and licensing review, was performed by Combustion Engineering (C-E) (Ref. 39). The evaluation presented herein is based on design information contained in this C-E study and follows to a large extent its safety and licensing review.

2.4.2 LICENSING ASPECTS REQUIRING FURTHER ATTENTION

This section lists the main potential licensing aspects that may require (in the opinion of C-E) more detailed evaluation for an HWR sited in the United States. Also presented by C-E for NRC review and comment are the proposed "applicant's positions," the information and arguments supporting their justification, as well as the resolution proposed for outstanding issues.

2.4.2.1 Use of Zirconium-Niobium Alloy and Rolled Joints in the Pressure Boundary of the Primary Heat-Transport System

The applicant's position is that the use of zirconium-niobium alloy and rolled joints can be shown to meet adequate safety standards if the following conditions are met:

1. At the standard CANDU-PHW pressure (1,600 psia):
   a. The Canadian data base on this subject (or a similar data base) is available, or
   b. An equivalent data base is developed in the United States.
2. At an increased pressure (2,250 psia):
   a. Condition 1.a or 1.b is met.
   b. A materials-development program is carried out to establish the strength and corrosion resistance of zirconium-niobium alloy at the new conditions.
   c. The rolled joints are shown to meet adequate safety requirements at the increased pressure.[a]

2.4.2.2 Low Probability and Minor Consequences of Pressure-Tube Failure

The applicant's positions are as follows:

1. Preservice and in-service inspection is adequate to insure that faults are detected well before they reach the critical crack size.
2. Pressure tubes have the leak-before-break characteristic, allowing early detection and ample time for remedial action and thus precluding more serious failures.

2.4.2.3 Limited In-Service Inspectability of Calandria and Calandria Tubes

The applicant's position can be summarized as follows: The calandria and calandria tubes are not pressurized and serve only a limited safety function. Hence, although the calandria and calandria tubes are designed and constructed as Class 1 components, they need not be subject to the full in-service inspection requirements of Class 1 components.

2.4.2.4 Potential for Small Positive Power-Reactivity Coefficient

The applicant's position is that the use of an automatic control system to stabilize the reactor is acceptable, because such a system

1. Is not required to be fast acting
2. Uses local in-core detectors
3. Is backed up by two fully independent protection systems, both using local in-core detectors

[a]Development of the nuclear sections of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code to cover the use of nonferrous material and rolled joints in the primary heat-transport system is not necessarily an NRC licensing requirement since the NRC may conclude, independently of the ASME, that these design features meet adequate safety standards.

4. Is not unique to HWRs for stabilization since it is already in use in, and has been licensed for, BWRs, which have a positive pressure reactivity coefficient, requiring continuous operation of an active pressure-control system for stabilization.

2.4.2.5 Presence of Xenon Instability

The applicant's position is that the use of an automatic control system to stabilize the reactor is acceptable, because such a system

1. Is not required to be fast acting
2. Uses local in-core detectors
3. Is backed up by two fully independent protection systems, both using local in-core detectors.

2.4.2.6 Presence of Positive Void-Reactivity Coefficient of the Coolant

The positive void-reactivity coefficient of the coolant is considered to be acceptable by the applicant for the following reasons:

1. There is a low probability of inserting significant amounts of reactivity because of the positive void coefficient, since this would require the occurrence of a loss-of-coolant accident.
2. The total reactivity inserted during the voiding of a single loop of the primary heat-transfer system is relatively small.
3. The mean neutron lifetime is long (about 10^{-3} second, i.e., about 30 times longer than for the LWR). The change in power is therefore much slower for a given reactivity insertion than it is for the LWR.
4. The HWR is equipped with two fast-acting, separate, independent, and diverse shutdown systems, each with enough reactivity worth to bring the reactor to the cold-shutdown condition.
5. Both shutdown systems are installed in the low-pressure moderator region and are not subject to the hydraulic forces associated with out-of-core loss-of-coolant accidents.

2.4.2.7 Potential for Limited Fuel Damage During Moderate-Frequency Events[a]

The applicant's position is that fuel damage (if any) is of economic concern only, for the following reasons:

1. Probability for fuel damage from moderate-frequency events is very low.
2. Failed fuel can be quickly detected and located, and easily removed, without requiring plant shutdown.
3. Site-boundary dose limits are not exceeded.

The HWR, operated with slightly enriched fuel, may have power-peaking factors associated with the fueling operation (fueling ripple) that are slightly larger than those of the CANDU-PHW reactor operated on natural uranium.

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[a]CANDU-PHW reactors operate with ample thermal-hydraulic margins for protection against fuel damage in moderate-frequency events, such as PHTS pump coastdown or loss-of-regulation incidents. This has been demonstrated by the satisfactory operation of the Pickering and Bruce nuclear power stations.

2.4.2.8 Use of Computers in the Automatic Reactor-Control System of the HWR

The applicant's position is that the use of computers in the automatic reactor-control system meets adequate safety standards because

1. Computer malfunction does not affect operation of the plant-protection system.
2. Control and protection functions are kept completely separate, even to the point of separate neutron-poison devices.
3. The use of computers has already been found to meet adequate safety standards in the protection systems of LWRs (Arkansas Nuclear 1-Unit 2).

2.4.2.9 Seismic Qualification of the Core, Calandria, and Fueling Machines

The HWR can be seismically qualified for most, if not all, U.S. seismic conditions[a] if (a) the results of the Canadian-Japanese analyses and experiments for the CANDU-PHW are made available and extended to the HWR conditions or (b) independent seismic analyses and experiments are performed in the United States for the HWR.

2.4.2.10 Performance of the HWR Emergency Core-Cooling System

The applicant's position is that the design-basis criteria for the HWR emergency core-cooling system need not include a maximum cladding temperature of 1,200°C (2,200°F) for the loss-of-coolant accident but should follow a time-at-temperature criterion (such as is used for the CANDU-PHW reactor) because of the following considerations:

1. Current U.S. criteria for the emergency core-cooling system (Appendix K, 10 CFR 50) were specifically developed for the closely packed cores of LWRs, to maintain a coolable core configuration and to prevent core meltdown.
2. The core of the HWR is less closely packed, and the emergency core-cooling system of the HWR is backed up by an independent, diverse, continuously operating core-heat-removal system (the moderator-cooling system), which is capable of preventing core meltdown, even in the event of ECCS failure (Refs. 49 and 50).
3. The time-at-temperature criterion, as applied in Canada for the CANDU-PHW reactor, is fully supported by experimental results, obtained in both Canada and in the United States, on oxygen embrittlement of the cladding (see also Section 2.2, Figure 2-2).
4. The highest cladding temperatures during a loss-of-coolant accident occur for break sizes that are about 20 to 30% of the maximum-size header break, when temporary flow stagnation takes place in the core region and when the primary heat-transport system in the core region is still close to its nominal operating pressure. Ballooning (or other deformation) of the cladding has a low probability under these circumstances.

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[a]As mentioned earlier, the core and pressure tubes of the current-design CANDU-PHW can, without modifications, withstand earthquakes with a ground (free-field) acceleration of 0.5g and higher, if provided with an adequate foundation (Ref. 21). The core and pressure tubes can withstand accelerations of 2 to 4g, measured inside the core, depending on the direction of the motion. For high seismic conditions, it is expected that some modifications may be required to the fueling machines, including possibly the installation of "soft snouts" (using bellows), as apparently already applied in the Japanese FUGEN reactor.

2.4.2.11 Tritium Release for HWRs

The tritium release rates for HWRs are expected to be well within the guidelines established by the ICRP for the following reasons:

1. Actual dose rates from tritium release are less than 1% of the ICRP permissible values for CANDU-PHW power plants.
2. Tritium dose rates for CANDU-PHW reactors could be further reduced (this is a cost/benefit question).

2.4.3 COMPARISON OF HWR AND CANDU-PHW CHARACTERISTICS WITH RECOMMENDED RESEARCH AREAS FOR IMPROVED LWR SAFETY

A recent NRC report to Congress on recommendations for research to improve the safety of light-water nuclear power plants (Ref. 51) identifies 16 research topics as shown below.

Plant Surveillance and Operation

1. Nondestructive examination and on-line monitoring
2. Improved plant controls
3. Improved in-plant accident response
4. Reduced occupational exposure

Safety Systems

5. Alternative emergency core-cooling concepts
6. Alternative decay-heat-removal concepts
7. Alternative containment concepts
8. Improved reactor shutdown systems
9. Reactor vessel rupture control
10. Core retention measures
11. Equipment for reducing radioactivity releases

Plant Configuration and Design

12. Advanced seismic designs
13. Improved plant layout and component protection
14. Protection against sabotage

Siting and Emergency Response

15. New siting concepts
16. Improved off-site emergency response planning

It should be noted that some of these topics have already been addressed for the HWR (or CANDU-PHW).

Research topic 2, "improved plant controls," is adequately addressed by the existing HWR design. These controls are developed as the first line of defense against events that could result in the release of radioactivity. The planned use of control systems for this purpose in the HWR provides the type of significant risk reduction being sought for LWRs under this research topic.

Research topic 3, "improved in-plant accident response," is partially addressed by the HWR and CANDU-PHW design. CANDU reactors are provided with a main control room and an auxiliary control room; the latter is installed at a separate location. Each control room has full capability for reactor shutdown to cold conditions and for providing continuous state-of-the-plant information. Light-water reactors also have this capability, provided that there is no equipment damage in either control room. In other words, the LWR is designed for conditions that would make the control room uninhabitable without damaging any of the equipment. However, in the CANDU design, this capability is maintained even if there is total failure of all equipment in the other control room. This feature would improve the in-plant accident response during an event such as a fire in one of the control rooms that would damage the control room equipment.

Research topic 5, "alternate emergency core-cooling concepts," is intended to decrease the risk associated with failure of current LWR emergency core-cooling systems. The HWR design provides redundant means of cooling the core under emergency conditions, namely through the emergency core-cooling system or through the moderator and its cooling systems. It is very likely that the HWR design meets the intent of the research topic currently being addressed for LWRs.

Research topic 6, "alternate decay-heat-removal concepts," is intended to improve the safety of the LWR by providing alternatives to the ultimate heat sink for decay-heat removal. It would appear that the CANDU-PWR design meets the intent of this research topic by using two independent cooling and emergency water supply systems.

Research topic 7 covers "alternate containment concepts." The dual-failure design approach used in CANDU, which demonstrates that the impact of impairment of the containment capability is acceptable, may meet the objective of this LWR research topic.

Research topic 8, "improved reactor shutdown systems," is more than adequately addressed by the HWR design. In fact, the HWR design approach goes even further than the intent of research topic 8 by considering the impact of the loss of each of the two independent shutdown systems separately, the independent impairment of containment capability separately, and the independent loss of the emergency core-cooling system separately.

Research topic 10 addresses the subject of "core retention measures." It is highly likely that the intent of this research topic has been met by the HWR and CANDU-PHW design, in that two independent emergency core-cooling systems are provided, which very well could reduce the probability of the core-melt event to the point that it requires no further consideration.

2.4.4 CONCLUSIONS

It is concluded that the HWR of the modified CANDU-PHW type, proposed by Combustion Engineering and discussed in the foregoing, is essentially licensable in the United States. This conclusion is consistent with the opinions of a number of independent sources (Refs. 39, 45-47).

Design changes have been incorporated into the HWR, relative to the CANDU-PHW, to be consistent with U.S. regulatory requirements. These include, for example, those design changes that were made to accommodate differences between the United States and Canada in the basic licensing assumptions underlying the seismic design.[a] Other design changes were incorporated to reflect current U.S. industrial practices.

A number of licensing aspects that appear to require further attention have been listed in Section 2.4.2. These are primarily issues concerning design features for which the U.S. available data base is as yet incomplete or for which an interpretation of the original intent of a current LWR licensing requirement would be necessary.

It is further concluded that, provided a suitable cooperative licensing arrangement can be made similar to those already entered into or planned between Canada (Atomic Energy of Canada Limited) and foreign countries for access to the Canadian data base, the first U.S. commercial HWR could be licensed and constructed on the same schedule as an LWR but preceded by a 3- to 5-year period for safety-related research and development (R&D) associated with changes to the Canadian design or verification of design required by U.S. licensing and construction practice. These R&D costs for the work described in Section 2.5 are estimated to be in the range of \$50 to \$250 million depending on the magnitude of the effort required to resolve the specific safety and licensing items that the NRC may identify as a result of their review of this document.

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[a]It should be noted that Canadian licensing requirements for seismic design are in full compliance with the Codes of Practice and Safety Guides of the International Atomic Energy Agency.

2.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Research, development, and demonstration requirements are related primarily to the differences in the design of the 1200-MWe C-E HWR and CANDU plants of similar size planned for construction in Canada. These differences include for example an increase in system pressure to reduce capital cost and improve thermal-cycle efficiency, and the use of enriched fuel to improve fuel utilization. Additional experimental data on the performance of slightly enriched uranium fuel may need to be developed by irradiating such fuel in existing HWRs (e.g., the Canadian NPD) to the discharge burnups anticipated for the reference design (about 21,000 MWd/MT).[a] Methods of analyzing the response of the HWR to anticipated operational occurrences and other postulated accidents may have to be developed and will have to be evaluated by the NRC, and an SAR in conformance with existing NRC criteria, or NRC criteria to be developed for HWRs, will have to be developed and defended. Additional requirements for licensing would involve experimental analysis of thermal-hydraulic performance for any fuel-bundle and fuel-channel designs developed in the United States.

The use of slightly enriched fuel and higher operating pressures should result in no fundamental change to HWR design but nevertheless will necessitate some development in order to accommodate the higher power peaking expected with slightly enriched fuels and the effect of higher system pressures on pressure-tube design and performance. Possible modifications for HWRs sited in the United States are somewhat difficult to quantify since a thorough licensing review of the HWR has yet to be completed.

The areas that have been identified as possibly requiring development effort to support the HWR design are briefly discussed below.

2.5.1 SAFETY-RELATED PHYSICS PARAMETERS

Lattice experiments in critical facilities may have to be performed for slightly enriched uranium fuel, to measure such parameters as core reactivity, coefficients of reactivity, power distribution, and control-absorber worth.

2.5.2 FUEL TESTING

Demonstration fuel assemblies, containing an initial enrichment sufficient to achieve the discharge burnups anticipated for the slightly enriched HWR, may have to be irradiated in test reactors.[a] These experiments would demonstrate the performance of such fuel up to the burnups anticipated and for power changes that occur during refueling.

2.5.3 MODIFIED PRESSURE-TUBE DESIGN

The reference NASAP HWR is expected to operate at pressures comparable to those of the PWR to increase net station efficiency and reduce the capital cost of the HWR. An experimental program to demonstrate the integrity of the pressure tubes and rolled joints at these higher operating pressures may have to be performed. This

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[a]This would probably be the case only if collapsible cladding were to be used. For free-standing cladding it is likely that enough irradiation experience is available from LWR fuel; furthermore, as noted before (Section 2.3.6.1), irradiation data up to 15,000 MWd/MTU does exist for current CANDU fuel (Ref. 40).

program might be similar in nature to tests already performed on CANDU pressure tubes and would be intended to extend the range of validity of experimental information to the higher pressures anticipated for the reference design.

2.5.4 THERMAL-HYDRAULIC TESTS

The empirical correlations used to establish the margin to burnout in CANDU reactors cover the range of pressures currently employed. These correlations will have to be extended to the higher pressures anticipated in the reference NASAP design.

2.5.5 FUEL CYCLE

Research and development related to the fuel cycle consists primarily of developing the experimental data base required for the design and licensing of HWRs operating on a slightly enriched uranium fuel cycle. This would include physics verification and the evaluation of irradiation behavior and fuel performance under transient conditions. Some of this information has been developed previously in the LWR program and is expected to be at least partially applicable.

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APPENDIX A

U.S. Nuclear Regulatory Commission Review of Safeguards Systems for the Nonproliferation Alternative Systems Assessment Program Alternative Fuel-Cycle Materials

BACKGROUND

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for nuclear material control and accounting. Requirements for the physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against attack, acts of sabotage, and theft. The U.S. Nuclear Regulatory Commission (NRC) has considered whether strengthened physical protection may be required as a matter of prudence (Ref. 1). Proposed upgraded regulatory requirements to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). A reference system described in the proposed upgraded rules is considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM SAFEGUARDS BASIS

The desired basis for the NRC review of safeguards systems for the Nonproliferation Alternative Systems Assessment Program (NASAP) alternative fuel-cycle materials containing significant quantities of strategic special nuclear material (SSNM),[a] greater than 5 formula kilograms,[b] during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed revisions cited above. The final version of the proposed physical protection upgrade rule for Category I[c] material is scheduled for Commission review and consideration in mid-April. This proposed rule is close to being published in effective form and, together with existing regulations, will provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base should be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U.S. authority, and where diversion by national or subnational forces may occur, proposals have been made to increase radioactivity of strategic special nuclear materials (SSNMs) that are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh-fuel material to require that, during the period after export from the United States and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover low-radioactivity SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time required in obtaining material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirements imposed by the Nuclear Non-Proliferation Act of 1978 include application of International Atomic Energy Authority (IAEA) material accountability

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[a] $\geq$20% U-235 in uranium, $\geq$12% U-233 in uranium, or plutonium.

[b] Formula grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium); Ref. 10 CFR 73.30.

[c] IAEA definitions of highly enriched uranium (>20%).

requirements to nuclear-related exports. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by the IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed alternative that could be used to provide additional safeguards protection against diversion of shipments of SSNM by subnational groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

NRC REVIEW

It is requested that NRC perform an evaluation of a spectrum of safeguards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. For the fuel cycles under review, consideration should be given to both unadulterated fuel materials and those to which added radioactive material purposely has been added. The relative effectiveness of various safeguards approaches (such as upgraded physical protection, improved material control and accountancy, dilution of SSNM, decreased transportation requirements, few sites handling SSNM, and increased material-handling requirements as applied to each fuel material type) should be assessed. The evaluation should consider, but not be limited to, such issues as the degree to which added radioactive contaminants provide protection against theft for bomb-making purposes; the relative impacts on domestic and on international safeguards; the impact of radioactive contaminants on detection for material control and accountability, measurement, and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tampering or breaching; the increased public exposure to health and safety risk from acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, the NRC must consider the export and import of SSNM as well as its domestic use.

As part of this evaluation, we request that the NRC assess the differences in the licensing requirements for the domestic facilities, transportation systems to the port of entry of the importer, and other export regulations for those unadulterated and adulterated fuel-cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, and on the international and national safeguards systems of typical importers for protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, as well as the potential advantages in detection or deterrence should be described in detail. The potential role, if any, that added radioactivity could or should play should be clearly identified, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its acquisition by foreign countries for weapons purposes. Licensability issues that must be addressed by research, development, and demonstration programs also should be identified.

Table A-1 presents a listing of unadulterated fuel materials and a candidate set of associated radiation levels for each that should be evaluated in terms of domestic use, import, and export:

Table A-1. Minimum radiation levels for various fuel material types

Fuel Material Type	Minimum radiation level during 2-year period, rem/hr at 1 meter (Ref. 6)	
	Mixed[a]	Mechanically attached[b]
PuO_2, $HEUO_2$ powder or pellets[c]	1,000/kgHM	10,000/kgHM
PUO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets[c]	100/kgHM	10,000/kgHM
LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[a]Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[b]Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[c]HEU is defined as containing 20% or more U-235 in uranium, 12% or more of U-233 in uranium, or mixtures of U-235 and U-233 in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Candidate methods and radiation levels are indicated in the following table and references.

Table A-2. Candidate methods and radiation levels for spiking fuel materials

Fuel material type	Minimum 2 year radiation level, rem/hr at 1 meter	Process	Minimum initial radiation level, rem/hr at 1 meter	Reference
a. PuO_2, HE UO_2 powder or pellets	1000/kgHM	Cobalt-60 addition	1300/kgHM	2, 3, 5, 6
b. PuO_2-UO_2 and HE UO_2/ThO_2 powder or pellets	100/kgHM	Cobalt-60 addition	130/kgHM	2, 3, 5, 6
		Fission product addition (Ruthenium-106)	400/kgHM	
c. LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	Co addition	13/assembly	2, 3, 5, 6
		Fission product addition (Ruthenium-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1000 (30 days)/ assembly	4
d. LMFBR or GCFR fuel assembly	10/assembly	Cobalt-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ruthenium-106)	40/assembly	2, 3, 5, 6
		pre-irradiation (40 MWd/MT)	1000 (30 days)/ assembly	4

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APPENDIX B

Responses to Comments by the U.S. Nuclear Regulatory Commission
PSEID, Volume II, Heavy-Water Reactors

Preface

This appendix contains comments and responses resulting from the U.S. Nuclear Regulatory Commission (NRC) review of the preliminary safety and environmental submittal of August 1978. It should be noted that the NRC comments are the result of reviews by individual staff members and do not necessarily reflect the position of the Commission as a whole.

RESPONSES TO GENERAL COMMENTS

1. Regarding the NRC request to reduce the number of reactor concepts and fuel-cycle variations, the Nonproliferation Alternative Systems Assessment Program (NASAP) set out to look at a wide variety of reactor concepts and fuel cycles with potential nonproliferation advantages. These various concepts have differing performance characteristics in other important respects, such as economics, resource efficiency, commercial potential, and safety and environmental features. The relative importance of these other characteristics and trade-offs has been determined and the findings are incorporated in the NASAP final report.

2. Regarding the comment on the need to address safeguards concepts and issues, some concepts for providing protection by increasing the level of radioactivity for weapons-usable materials have been described in Appendix A to each preliminary safety and environmental information document (PSEID). Appendix A has been revised to reflect NRC comments.

   An overall assessment of nonproliferation issues and alternatives for increasing proliferation resistance is provided in Volume II of the NASAP final report and reference classified contractor reports.

INTRODUCTION

Upon reviewing Vol. II on heavy-water reactors (HWRs), the U.S. Nuclear Regulatory Commission (NRC) requested Brookhaven National Laboratory (BNL) to review sections 4 and 5 and assess the potential licensing issues for the Nonproliferation Alternative Systems Assessment Program (NASAP) heavy-water reactor.

The BNL reviewed sections 4 and 5 and prepared a set of questions which was transmitted to the NRC. In addition, several questions were raised directly with Combustion Engineering, Inc. (C-E).

The NRC prepared its questions taking into consideration the concerns of BNL.

Part I of Appendix B responds to NRC comments, Part II includes the responses to specific BNL questions, and Part III includes the C-E responses made directly to BNL questions.

PART I

RESPONSES TO U.S. NUCLEAR REGULATORY COMMISSION COMMENTS

Question 1

Natural convection. Although the C-E design may not be sufficiently detailed to assess its potential for natural circulation decay heat removal, are there specific design steps that could be taken to augment natural circulation? In view of the possibility of steam bubbles in the horizontal pressure tubes, are there reasons (experiments) to believe that natural circulation would not be inherently ineffective in this type of reactor?

Response

A detailed evaluation of the available natural circulation heat removal capabilities of the C-E heavy-water reactor (HWR) design has not been undertaken. A rough estimation of the available capacity for heat removal by natural circulation indicates, however, that adequate heat removal capability would be available to remove the decay heat that would exist following reactor coolant pump coastdown assuming the reactor tripped from 100% power at the same time that all four coolant pumps lost power. This estimate is conservative in that subcooled heavy water was assumed when in fact saturated water could exist.

It might also be noted that HWRs of the Canadian design have a natural convection cooling capability, and that this mode of cooling has been demonstrated on operating CANDU units. Although a detailed assessment of the natural convection cooling capability of the C-E design would have to be performed to establish definitively its potential for natural circulation decay heat removal, there are no obvious design differences with respect to CANDU plants which would appear to compromise the natural circulation decay heat removal capability demonstrated for operating CANDU units.

Analysis and experiment on CANDU units indicates that the natural circulation heat removal capacity is essentially proportional to the decay heat source; i.e., a 4% of full power decay heat source results in a sufficient hot-leg to cold-leg pressure head, due to differences in coolant density, to effect a natural circulation coolant flow of approximately 4% full power flow. Under normal conditions, the formation of steam bubbles (in excess of the quality normally present at the pressure tube outlet) is not anticipated. However, should such steam bubbles form, they would begin to form toward the outlet of the pressure tubes (where coolant temperature is highest) and would be swept into the outlet header by the coolant flow (i.e. would result in an increase in exit quality). Since the presence of these voids in the hot leg would increase the hot-leg to cold-leg pressure head, the presence of these steam bubbles would serve to increase the natural circulation flow rate. The higher flow rate caused by sweeping out steam bubbles into the hot leg would subsequently serve to quench the steam bubbles in the pressure tubes, and once quenched the natural circulation rate would return to its unenhanced value. If the decay heat source remains sufficiently high at the unenhanced natural circulation flow rate, steam bubbles may reform. This can lead to a cyclic behavior in which the formation of steam bubbles enhances natural circulation flow, which subsequently leads to the collapse of steam bubbles and unenhanced natural circulation flow, which then leads to the reformation of steam bubbles with increased natural circulation flow, etc. This type of cyclic behavior has been demonstrated on operating CANDU plants.

Question 2

Primary heat transport system. We share the concern expressed by BNL[a] that the two primary loops are connected at a common pressurizer, though they are otherwise independent. Because of the reliance placed on isolation of these loops to maintain a loss-of-coolant reactivity less than one dollar, discussion along the lines suggested by BNL would be useful. How reliable are the pressurizer isolation valves against improper activation?

Response

As this comment notes, the conceptual HWR contains two separate coolant loops each containing two steam generators, two pumps, and one-half of the reactor fuel channels and end connections. The only connection between these two loops is the surge line that connects each loop to a common pressurizer. To ensure that in the event of a rupture of one of the loops the intact loop does not blow down through this common connection, the conceptual HWR design contains two redundant isolation valves in each of the surge lines connecting each reactor coolant system (RCS) loop header with the pressurizer. These isolation valves are illustrated in Figure B-1 (Figure 5.1.3-7, Ref. 1). These valves are conceived as being air operated with a safety-grade air source and fail close upon loss of air or electrical power. Each valve in the affected loop will receive a different signal to close in the event that a loss-of-coolant accident (LOCA) event occurs. This signal will be based on low reactor coolant header pressure in conjunction with high containment pressure. The exact signal generation basis will depend upon detailed accident analyses which have not been conducted. Emergency power and safety grade air sources will ensure that the valves in the intact loop will remain open, or be reopened, by the operator

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[a]Responses to Brookhaven National Laboratory comments and questions are given separately after the responses for the NRC comments and questions.

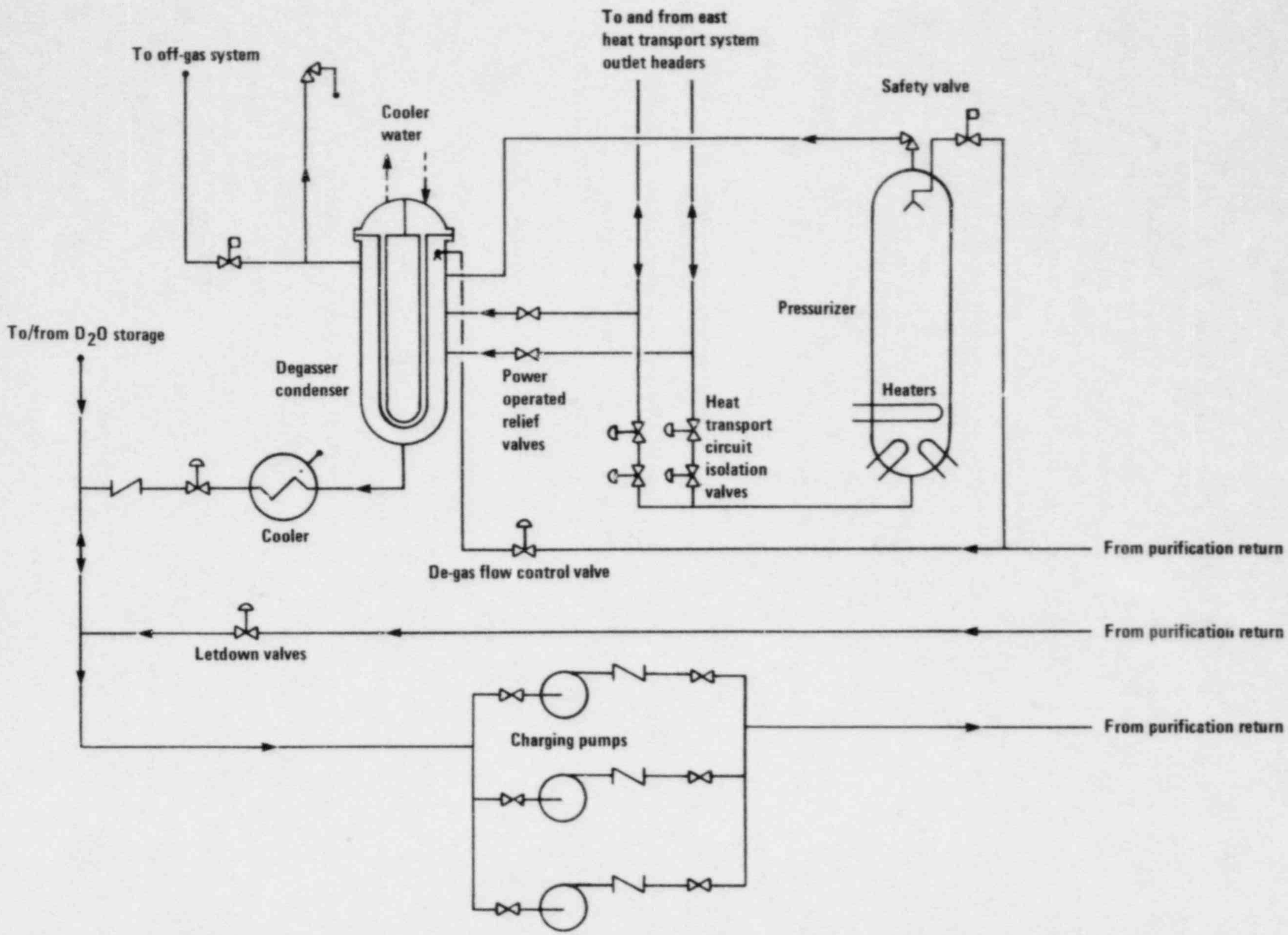

Figure B-1. P.H.T.S. pressure and inventory control system.

to provide pressure control during cooldown of the intact loop. The closure time of these valves will be specified as a maximum of 5 seconds. Over-pressure protection for the RCS is provided by relief valves on the reactor side of these valves. The use of redundant isolation valves should provide a high degree of assurance that the RCS loops are isolated in the event of a LOCA event.

It should also be noted that the isolation of the two loops is not strictly necessary from a safety standpoint, although, of course, if isolation fails to occur, the reactivity inserted during the loss-of-coolant accident is somewhat greater; in fact, the entire RCS is interconnected at the Bruce Station, so that the entire RCS will blow down in the event of a LOCA event in this plant. As noted in the attachment (BNL letter), the amount of reactivity inserted due to complete voiding of an RCS loop is substantial and the complete voiding of both loops would insert more than one dollar. These amounts of reactivity are not inserted instantaneously, however, because it takes some time for the coolant channels to void; this is particularly true of the intact loop because blowdown must occur through the pressurizer surge line. The amount of reactivity insertion due to fuel channel voiding thus increases slowly as the blowdown progresses; hence, the initial periods are much smaller than that indicated in Table 7 of the BNL attachment, even if both coolant loops should blow down. Since the power transient is terminated by reactor scram, prompt criticality is never reached in the transient, and this is thought to be true even if the intact loop fails to isolate.

Question 3

Moderator cooling. The BNL reference to the PWR plena as heat sinks is well taken although the calandria vessel is certainly much larger and cooler. We also note in Nuclear Engineering International, January 1979, the article by J. T. Rogers describing calculated heat transfer to the moderator by use of the codes IMPECC and CONCYL. Are you aware of any experimental verifications of these codes, or do you think such verification would be feasible? Would it be a suitable subject for future research?

Response

The existence of any prior experimental verification of cooling via heat transfer to the moderator is not known. Although experimental verification would be difficult, it should, nevertheless, be possible.

It should be noted that the safety performance of the conceptual design is not predicated upon such moderator cooling. Moderator cooling is necessary only in the instance of the emergency core-cooling system (ECCS) failure. Since the ECCS of the conceptual reactor is designed to the same safety criteria as those employed in LWRs, a failure of this system is extremely unlikely and, as is the case for the LWR, need not be predicated. Moderator cooling is thus a line of defense, which goes one step beyond that available in the LWR, to provide additional heat removal in the very unlikely event of ECCS failure.

Question 4

A loss-of-heat-sink scenario. What analysis or experiments cover a two-phase flow situation in the horizontal tube geometry? What would be the effect of bubbles on natural convection circulation?

Response

To ensure that each loop will have cooling by a minimum of one steam generator, each steam generator in the conceptual HWR design is provided with two redundant atmospheric dump valves. Each dump valve will be capable of removing decay heat, reactor coolant pump heat, and cooling one loop down to shutdown cooling initiation temperature and pressure. No single active failure will prevent operation of at least one dump valve per steam generator. Main steam safety valves will dissipate the system energy if the steam pressure reaches their set point prior to opening the dump valves. Makeup water to the steam generator is ensured by the use of redundant emergency feedwater supplies that meet single-failure criteria.

As noted in the response to Question 1, the formation of bubbles is expected to enhance the rate of natural convection circulation.

The above discussion has assumed that emergency feedwater would be available for decay heat removal via the steam generator atmospheric dump valves. Due to the redundancy of emergency feedwater supplies, the availability of emergency feedwater is typically assumed in LWR loss-of-heat-sink accident analyses. However, the BNL attachment (Question 4) indicates that this question may be directed more at a Three Mile Island (TMI) type accident (where the emergency feedwater system did not operate because of valve misalignments), thus resulting (at least for a period of time) in a complete loss of the secondary side heat sink. Although a TMI-type accident analysis has not been performed for the conceptual HWR design, a number of qualitative comments can be made. First, the conceptual HWR design employs U-tube steam generators which contain a substantial secondary water inventory. This secondary water inventory would allow approximately 20 to 30 minutes to establish emergency feedwater before the steam generators went dry. Secondly, the continued loss of coolant from a stuck pressurizer (or header) relief valve would not occur in the HWR design since these valves vent into the degasser-condenser system (Figure B-1) which is designed to accommodate full system pressure. As the degasser-condenser fills, back pressure will prevent the continued discharge of primary coolant through the stuck relief valve. (Overpressure protection is provided by a safety valve on the degasser-condenser; this valve would open only if system pressure continued to rise, but would not normally open on the brief overpressure transient which occurred at TMI.) Lastly, power-operated relief valves are provided on the outlet header. Bubbles which may be formed within the fuel channels and which might accumulate in the outlet headers could thus be vented from the system through these valves, so that there would be no need to utilize the pressurizer for this purpose. In the event of a continued and complete loss of the secondary as a heat sink, ECCS injection, coupled with discharge from the power-operated relief valves on the outlet headers, would provide a means of fuel cooling.

In view of the above discussion, there appears to be no further need for an additional heat sink, such as the use of moderator inventory as suggested in the BNL letter. This alternate heat sink is, of course, available, but becomes effective only after considerable fuel and structural damage has occurred; consequently, no credit for this feature has been taken in any C-E analyses, and credit is taken for the moderator as an alternate heat sink in Canadian analyses only in conjunction with the assumed failure of the emergency core cooling system. In particular, the gas-filled space between the pressure tube and calandria tube serves as an effective insulation as long as pressure-tube geometry is maintained. In the event of failure of the ECC system, the severe overheating of the fuel and subsequently of the pressure tube will cause the pressure tube to lose strength and sag to the point where it will contact the calandria tube; it is only after contact that significant heat transfer to the moderator occurs. The Canadian analyses indicate that the heat transfer in this geometric configuration is sufficient to prevent meltdown from occurring. Clearly, significant fuel damage and structural damage to the pressure tubes will have occurred.and so the use of the moderator as an alternate heat sink is not viewed as a normal mitigating system; rather, it is considered an inherent feature of the pressure tube HWR design, which can provide some assurance that meltdown will not occur even in the highly unlikely event of failure of other engineered systems.

It might also be noted, for the record, that operating CANDU plants employ a full pressure shutdown cooling system and also have provisions for diverting emergency core cooling water to the steam generators to provide an alternate source of feedwater. However, consistent with LWR design practices, these features have not been incorporated into the C-E conceptual design.

Question 5

Common mode heat removal failure at headers. Are there break locations such as that suggested by BNL where the emergency core-cooling system would fail to cool a substantial part of the core? How successfully does flow reversal work if the ECCS must be switched to the outlet header?

Response

The conceptual HWR design incorporates ECCS cooling to both the inlet and outlet headers. Upon actuation, emergency core cooling is simultaneously injected into both inlet and outlet headers, and there is no need to "switch" emergency core cooling flow to the outlet header. Although a detailed quantitative evaluation of emergency core cooling has not been performed, qualitatively the system would appear to be quite satisfactory. While the coolant injected into the inlet header would flow out the break in the event of an inlet header break, coolant injected into the outlet header would have to flow past the fuel (in the reverse direction) before it could exist through the break, thus providing considerably better fuel cooling than would occur if only inlet injection were employed.

Question 6

Temperature and void coefficients. The BNL calculation of temperature coefficients is admittedly very preliminary; nevertheless, the indicated trends should be pursued further.

The temperature coefficients apparently become more positive or less negative as burnup proceeds. The moderator temperature coefficient appears to be slightly positive at equilibrium burnup. We are told that the trend of these coefficients with burnup corroborates Canadian calculations. What is the effect of these positive coefficents on kinetic behavior at power? Are there any instabilities?

Although we do not necessarily endorse the view of BNL that "there appears to be entirely too much reactivity associated with voiding one loop..." we are interested to know if alternatives to the two-loop design have been considered from the physics point of view. The designers have rejected the feasibility of dividing the core into more than two independent loops on the basis of capital cost. What are the maximum period limitations to offset the capital costs and how much cost is involved?

Response

The potentially slightly positive coefficients do not appear to adversely affect the kinetic behavior of the reactor at power or to create instabilities. As the BNL (and C-E) calculations indicate, these coefficients are only very slightly positive. A zonal control system that introduces or removes light water from control chambers to adjust local power densities is also provided. The successful operation and high availability of operating CANDU units would seem to testify that these coefficients do not result in operational problems.

As noted in the response to Question 2, the calculated periods that would occur upon complete voiding of a coolant channel are an insufficient indicator of the effect of reactivity insertion during the LOCA event. As mentioned previously, the actual rate of power increase during the LOCA event will be much slower than these calculated periods because of the time required to reduce system pressure and cause voiding by venting through the break; the reactivity transient is terminated by reactor trip well before such periods are reached. It is possible that the periods obtained with either the two- or three-loop configuration are acceptable; hence, period considerations provide no basis for choosing between the two alternatives.

Question 7

Xenon oscillations. It appears that this problem is being addressed in the C-E design. Are allowances being made for abnormal behavior and for the increased complexity of the larger system?

Response

In general, allowances have been made for the larger system by providing increased in-core detector coverage and power distribution control elements. Clearly, detailed design analyses of xenon stability would have to be performed as part of the control system development program for a large plant design.

Question 8

Neutron behavior associated with a LOCA event. Has neutron streaming in voided or partially voided horizontal tubes been estimated? If the upper one-third to one-half of a tube is voided, does this provide a direct path for well-thermalized neutrons to reach the center fuel rods that normally do not see as much thermal flux? What effects would this have on reactivity coefficients?

Response

Evaluations of the effect of neutron streaming during LOCA events have not been performed and it is not known what the effects would be on reactivity coefficients. Although such streaming might lead to somewhat more positive coefficients, the effect of uniform voiding would seem to be similar because this also allows thermalized neutrons to reach the center fuel rods more readily; increased streaming along the length of the fuel rods will, of course, lead to a more negative coefficient. On balance, one would expect little effect, although detailed calculations would certainly be needed to verify this supposition.

Question 9

References 15 and 18 are 7 years old. Is more data available regarding irradiated properties of zirconium-2.5% niobium? (Fracture toughness?)

Response

Additional references can be found on page 4.9-6 of Reference 1.

Question 10

Since the pressure tube is part of the reactor coolant boundary, it would be desirable to present both the material and tube joint as an approved American Society of Mechanical Engineers (ASME) Code Case for a licensed reactor. Assuming that the only data available are those already in the public domain, what is your estimate of the time and resources required to get a code case ruling on the material and the tube joint?

Response

It is estimated that it would take approximately 1 year to prepare a code case for the Zircaloy alloy pressure tube and pressure tube joint. Experience with the time required to get such a code case approved by the ASME has varied considerably; some code cases have been approved within a 3- to 6-month period, while others have taken 7 years or longer. The length of time required to obtain code case approval depends partially on the quality and extent of available information, and partially upon the urgency and incentives for code case development. Experience has shown that requests for code cases that are either academic in nature or that benefit a single vendor are apt to be deferred in favor of more pressing requests for code cases, and are apt to take long periods for approval. On the other hand, ASME attention has focused on those requests where the incentives were high and where there had been widespread vendor and NRC interest; in this situation, fairly rapid responses have often been obtained.

Question 11

Discuss why you believe linear elastic fracture mechanics is an appropriate tool for assessment of the leak-before-break possibility, which may be sominated by an aggressive environment such as stress corrosion or radiation damage.

Response

The assessment of the leak-before-break phenomena has not been based upon linear elastic fracture mechanics, but rather upon elastic-plastic fracture mechanics. There appears to be ample experimental information to justify this type of analysis, such as that reviewed in Section 4.2.4--Pressure Tube Design Considerations (Ref. 1).

Question 12

Postulating that the leak-before-break hypothesis can be satisfactorily demonstrated, and that sufficient time exists for leak detection and reactor shutdown before a self-propagating crack develops, discuss whether the leak-detection system should be considered a safety grade system, Seismic I, Single failure, etc.

Response

The annulus gas system (described more fully in Section 5.1.2.5 of Ref. 1) is designed to Seismic Category I--Safety Class 3 Standards. These categories are normally appropriate to a monitoring system where possible failure does not lead to a violation of safety limits and where possible failures can be detected and corrected.

Question 13

Can reasonable assurances be given, a priori, that the risk associated with rupture of the primary coolant boundary of the HWR is comparable to the risk of pressure vessel failure for an LWR? What in-service inspection procedures, parallel materials research studies, and engineered safety features are included in the program to assure that the risk of pressure tube failure plus failure propagation in the HWR is as low as that of pressure vessel failure in the LWR?

We believe that it would be helpful in making a judgment of licensability if the following subjects were addressed:

Section 2.4.2.2. Expand to give scope of in-service inspection program and materials research. We are not persuaded that questions of failure propagation due to pipe whip and missiles have been satisfactorily resolved, especially at the embrittled material conditions at end-of-life.

Section 2.2.2. Can the entire pressure tube be inspected for cracks without unloading the fuel or just the region near the rolled joint? If the moisture detection of leaks-before-breaks is determined to be not sufficiently reliable, what frequency of direct UT or acoustic inspection for cracks would be necessary as a supplement? Is it feasible to perform such inspections with this frequency?

Response

It is not clear precisely what is meant by "risk associated with rupture of the primary coolant boundary of the HWR." If what is meant is the probability of pressure tube rupture, then it is reasonable to assume that the "risk associated with the rupture of the primary coolant boundary of the HWR" is comparable to the risk of pressure vessel failure for an LWR. While the probability of a catastrophic pressure tube failure might be judged comparable to the failure probability of a pressure vessel, the overall probability of a rupture of the primary coolant boundary of the HWR is probably significantly higher than the failure probability of an LWR vessel because of the large number of pressure tubes present in the HWR. However, "risk" should not be considered synonymous with failure probability, but rather with failure probability times consequence of failure. In the case of an LWR, pressure vessel failure is extremely serious, and for sufficiently large failures at certain locations, could be postulated to lead to "Class 9" consequences involving fuel melt. In the HWR, the consequences of a pressure tube failure (or even of the failure of several pressure tubes) are nowhere near as severe as the consequences of an LWR vessel failure; in fact, the consequence of a pressure tube failure is less severe than that of the design-basis LOCA event. Pressure tube rupture is not particularly severe since the coolant flow through the ruptured pressure tube is limited by the diameter (and pressure drop) of the feeder tubes that connect the pressure tubes to the inlet and outlet headers. These feeder tubes have a maximum diameter of 3.4 inches, thus severely limiting blowdown through a ruptured pressure tube. Pressure tube rupture is considered in the Canadian safety evaluation, where it is found to be acceptable (nonlimiting).

With respect to failure propagation, it is difficult to quantify the probability of such failure propagation. However, as noted in response to Question 15 below, experiments have been performed on defective tubes that indicate that the failure of a pressure tube will not compromise the integrity of the neighboring pressure tube, thus precluding failure propagation (which involves the sequential failure of a number of pressure tubes). Based upon experiments such as this, the Canadians have concluded that the probability of failure propagation is insignificant, and there seems to be no evidence to contradict this conclusion.

The considerations mentioned above lead us to the conclusion that the risk (probability times consequence) of a rupture of the primary coolant boundary of the HWR is comparable to the risk (probability times consequence) of pressure vessel failure for an LWR.

A development program aimed eventually at the commercial deployment of the conceptual HWR design would have to include materials and failure propagation studies designed to extend available information and experiment into the range of primary temperature and pressure postulated for the conceptual design. In particular, this program would develop information on the material properties and fracture mechanics of irradiated pressure tube material operated at the higher temperatures and pressures postulated for the conceptual design. Extension of the fuel propagation experiments (as discussed in Question 14) to the pressures of the conceptual HWR design should also be performed.

Using available technology, the entire fuel tube including the rolled joint can be inspected using ultrasonic techniques. Inspection is currently performed by a probe inserted by the refueling machine (in place of fuel) and can currently be performed only when the reactor is shut down. Advanced inspection techniques are under development in Canada which might eventually allow inspection at power. Canadian inspection programs employ a spot-check approach aimed at verifying the conditions of the pressure tubes and at detecting any developing problems before they can lead to serious pressure-tube failures.

U.S. licensing requirements for Class 1 structures impose somewhat more formalized inspection requirements which will probably necessitate systematic inspection of the pressure tubes (rather than the Canadian spot-checking). The frequency of such an in-service inspection program would be such as to ensure that any large fault which forms or is introduced into the pressure tube cannot penetrate the pressure tube during the period between in-service inspections. Of primary concern here is the possibility of introducing faults which approach or exceed the critical crack length. (One suggested method, for example, of introducing such a fault might be during refueling if a tool were inadvertently left in the fueling channel.) Such an inspection program would thus ensure that large faults were not present, in which case any small or incipient faults which penetrated the pressure tube would begin to leak well before the pressure tube ruptured. The frequency of in-service inspection depends both upon the maximum credible fault which can be introduced and upon the number of thermal cycles to which the pressure tube is subjected during the period. Although the period of in-service inspection will depend on the detailed system design (the force which the refueling machine can exert, pressure tube thickness), Canadian and British evaluations indicate that it will take a period of at least 10 years for such an introduced fault to penetrate the pressure tube; an in-service inspection program which consequently serves to inspect each pressure tube once every 10 years would be sufficient.

If the moisture detection of leaks-before-breaks is determined to be not sufficiently reliable, the frequency of in-service inspection would have to be increased or improved in-service inspection techniques would have to be developed. Such an in-service inspection program would have to be capable of detecting smaller faults than is currently thought possible with current inspection technology, or the inspection frequency would have to be decreased so that faults well below the critical crack length could be detected. It is difficult to estimate the required frequency of inspection using existing technology, but it is probably too frequent to be considered feasible. The development of on-line inspection techniques would allow frequent inspection, but it is speculative as to whether small faults could be detected under on-line service conditions. Another alternative under development is acoustic listening; these techniques would listen for pressure tube leaks, as an alternative to moisture detection.

In summary, existing in-service inspection techniques appear adequate in view of the well-established fracture mechanics of pressure tubes, the demonstrated ability to detect leaks (for example on the Pickering reactor), and the modest consequences of a pressure tube failure. There is always an incentive for improved in-service inspection, however, and such improved inspection techniques are under development in Canada. Improved in-service techniques should also be developed as part of a research and development (R&D) program aimed at the eventual deployment of an HWR in the United States, although the successful development of such techniques is not considered to be a prerequisite for such deployment.

Question 14

Have experiments or analyses been performed with respect to jet impingement or tube whip against the calandria tube and, if so, what conclusions were reached?

Response

A number of experiments have been performed to establish the effect of pressure tube failure on the calandria tube and upon neighboring fuel tubes. One set of experiments, reported in Reference 3, employed defective pressure tubes to establish the effect of pressure tube failure on the surrounding calandria tube (i.e., the effects of jet impingement and tube whip). The experiments found that the initial shock from discharge of coolant through the defect would not cause the calandria tube to fail (i.e. the calandria tube would not fail due to jet impingement or pipe whip). As coolant continues to discharge from the defective pressure tube, the pressure in the calandria tube reaches a peak within about one-half second. Whether the calandria tube can contain the coolant or not is dependent on the calandria tube strength, and pressure buildup, which, in turn, is dependent on coolant conditions and the exhaust area from the annulus through the end fitting and end shield assemblies. The test results and analysis indicated that by increasing the exhaust area, the possibility of calandria tube rupture can be reduced. In the standard CANDU design, however, it appears difficult to guarantee that the calandria tube will not rupture as a result of overpressurization, and so no assumption of calandria integrity is used in Canadian safety analyses. The experiments do indicate that impingement or pipe whip will not fail the calandria tube and the failure occurs only as a result of overpressurization.

The second series of experiments, reported in Reference 2, were aimed at establishing the effect of a pressure tube rupture on neighboring fuel channels. In these experiments, no calandria tube was employed around the intentionally defective pressure tube, so as to establish an upper limit on the consequence of a pressure tube burst on the surrounding fuel channels (which contain a pressure tube surrounded by a calandria tube). These experiments indicated that the explosive failure of a pressure tube would not cause the neighboring calandria (or pressure tube) to fail. Damage appeared limited to some deformation and denting of the target calandria tube. It was concluded that ". . . pressure tube rupture propagation from one channel to the adjacent one is very unlikely. . . . In addition, the dynamic loads in core structures caused by the pressure peaks in the D_2O tank, do not produce any relevant damage to the main structural functions."

Question 15

If collectively the pressure tubes are to have an equivalent reliability as a BWR or PWR vessel, then even greater reliability of the individual pressure tubes is required in the HWR. What is this estimated greater reliability and is it demonstrable?

Response

As noted in response to Question 13, because the consequences of a BWR or pressurized-water reactor (PWR) vessel failure are potentially more severe than the consequences of an HWR pressure tube failure, it cannot be concluded that an equivalent degree of reliability is necessary. For consequences comparable to that of an LWR vessel failure, pressure tube propagation would have to occur, and hence one would have to establish the probability of such failure propagation in order to establish the reliability requirements of individual pressure tubes. Such analyses are well beyond the scope and detail of the U.S. evaluation, although they may perhaps have been performed in Canada. One of the major difficulties in establishing this reliability estimate is obtaining reasonable values for estimates of pressure tube failure propagation. Since experiments to date have failed to demonstrate failure propagation (see response to Question 14), this probability is thought to be exceedingly low.

Question 16

Discuss how comparability with Appendix K would be demonstrated with equivalent margins of safety. What R&D may be needed to show comparable safety with respect to blowdown, metal water reactor, reflood, and post accident heat removal?

Response

As noted in the PSEID, a primary difference of significance between the Canadian and the 10 CFR 50, Appendix K, LOCA analysis is in the peak allowable clad temperatures; rather than employing an absolute temperature limit (1,200°C) as part of the oxygen embrittlement criteria, Canadian licensing employs a time-temperature relationship. Since there appears to be a well-established technical basis for this time-temperature criteria, the NRC, after detailed review, may find this approach acceptable. Alternately, HWR power density could be decreased (by adding additional fuel channels) so that the U.S. Appendix K limits are not exceeded. (Peak temperatures occur during the blowdown phase of the accident analysis; therefore, ECCS performance will not influence the peak temperatures obtained.) Other aspects of Appendix K with respect to blowdown, reflood, etc.. do not appear to be strictly applicable to the pressure tube geometry of the HWR. Consequently, equivalent assumptions will have to be developed and approved by the NRC. The other difference between Appendix K and the Canadian LOCA analysis noted in the PSEID is the explicit calculation in Canadian licensing practice of fuel failures and activity release. This aspect of the Canadian analysis is derived from its requirement to evaluate the LOCA event in conjunction with assumed impairment of containment. Imposition of Appendix K specified releases would consequently be no problem in the context of U.S. licensing requirements where containment impairment is not assumed.

In order to assess definitively the comparability and the equivalent margins of safety with respect to Appendix K, a detailed LOCA analysis will be required, employing applicable Appendix K assumptions. As part of this evaluation, equivalent assumptions will have to be developed for those Appendix K items that are not applicable due to the pressure tube configuration of the HWR. Research and development may be required to fully develop the time-temperature criteria for U.S. application, as well as for establishing approved evaluation models for blowdown, reflood, etc.

Question 17

Section 2.2.2 (p. 2-9). An amplified discussion of the means of protection against failure of the automatic control systems is required.

Response

The means of protection against failure of the automatic control system is discussed in greater detail in Section 5.5.6 of Reference 1. Basically, the HWR is provided with a protection system which utilizes ex-core detectors to provide trip signals on high neutron power and rate of change of neutron power, and in-core detectors to provide trip signals on high local power density. In the event of a (complete) failure of the automatic control system, trip signals are generated when either core power level, rate of change in power level, or local power density exceeds the trip setpoints. This is, of course, analogous to the response of the LWR in the event of a control system malfunction. Two separate shutdown systems are provided (shutdown rods and poison injection), each with independent and redundant trip signals. In the event of loss of such control functions as zonal control, the control system would function to first reduce core power in a controlled rampdown or effect a more rapid reduction in power through the dropping of control rods so as to avoid the necessity of a reactor trip; should these control actions fail, reactor trip would, of course, occur.

Question 18

Section 2.2.3. (p. 2-11). The statement is made that "A serious fault in the process system is defined as one that would, in the absence of safety systems, result in a substantial release of radioactive material to the environment." Later on the same page, under item 2, the statement is made that "a serious failure is one that in the absence of protective action would lead to serious fuel failure."

Are these statements in conflict? In connection with items 5 and 6 on the same page, please note that U.S. licensing will require conformance to applicable sections of the U.S. Code of Federal Regulations (CFR) and Regulatory Guides in regard to acceptable levels of effluents from normal operation and accidents.

Response

Properly, a serious fault in the process system is defined as one that would, in the absence of safety systems, result in a substantial release of radioactive material to the environment. Fuel failure would typically accompany such a serious fault, since it is only with fuel failure that substantial quantities of radioactive material are available for release, but fuel failure in itself is not considered a violation of Canadian safety limits. With respect to items 5 and 6 on pages 2 through 11 of the PSEID, allowable releases appear consistent with the CFR. Note that item 6 imposes requirements on the effectiveness of the safety systems for all faults (including limiting or design basis faults) which are equivalent to the releases allowed under the CFR for anticipated operational occurrences. Releases allowed under the CFR for limiting faults are acceptable under the Canadian code only for dual failures in which the complete failure of the mitigating safety system must be assumed.

Question 19

Page 2-16. The NRC would be inclined to continue the use of the source term defined in NRC Regulatory Guide 1.3 unless inherent differences between LWRs and HWRs provide a substantial basis for expecting considerably different accident behavior in the HWR. In this connection, we would consider the burnup, gap pressure, clad design, ECC temperatures, and any other notable differences between the reactor systems. If these can be shown to effect considerable reduction in the source term with a high degree of assurance, consideration would be given to appropriate modifications of the source.

The Canadian practice, as described here, appears to be similar to the more realistic calculations of the source term, as done in WASH-1400, rather than the conservative calculations that the U.S. licensing procedure uses. It would be inconsistent with our review of LWRs to calculate the HWR source term in this way without first showing major differences in the scenarios.

Please submit any such discussion of major scenario differences that you believe to be relevant.

Response

As mentioned briefly in response to Question 16, Canadian licensing requires the evaluation of the LOCA in conjunction with an assumed impairment of containment (dual failure). Because of the assumed containment impairment, a direct pathway is available for release of radioactive material to the environment in a LOCA event. This necessitates the explicit calculation of fuel failures during the LOCA event and of source terms in order to ensure that the evaluated site boundary doses do not exceed licensing limits (which are equivalent to our limits for limiting faults). Under U.S. licensing, it is not necessary to assume containment impairment during the LOCA event. Without this assumed pathway for release of radioactive material to the environment, the more realistic source term calculation employed by the Canadians would not be necessary, and acceptable consequences should be obtained using the NRC Regulation Guide 1.3 source terms.

Question 20

Is it justifiable to assume that the safety analyses, if carried out, would lead to results comparable to light-water reactors? In what areas do you foresee major differences?

Response

A more detailed discussion of this issue can be found in Section 5.5 of Ref. 1. As noted in this section, Canadian licensing imposes more severe assumptions than does U.S. licensing (i.e. dual failures, in which the unavailability of the mitigating safety system must be assumed). These dual failure events are limited to the same consequences (site boundary doses) as are U.S. limiting faults (where the dual failure is not assumed). As a result, more realistic calculations (for example the source terms described in response to Question 19) are often employed. It is difficult to trade the effect of the more demanding Canadian failure assumptions with the often increased conservatism present in U.S. calculations. However, we feel on balance that the results of a safety analysis would be comparable to light-water reactors, although clearly the performance of a detailed quantitative safety analysis for the HWR using U.S. assumptions would be required to confirm this supposition.

Areas of major difference are highlighted in Section 5.5 of Ref. 1. The most significant items are listed below:

1. Pressure tube as primary system pressure boundary
2. Power and void coefficients of reactivity
3. Peak clad temperatures in the loss-of-coolant accident
4. Relationship of fuel damage to safety and licensing

Since the first three of these items have been discussed in some detail in response to other questions, only the last needs some amplification. This item is also discussed in greater detail in Section 5.5.3.4 of Ref. 1.

Under U.S. code, fuel damage is not acceptable for anticipated operational occurrences and the amount of fuel damage is also limited for faults of moderate frequency. Under Canadian licensing, there is no explict prohibition against fuel damage during such faults. In practice, sufficient margin is provided to trip set points to ensure that fuel damage does not occur during anticipated events, because fuel failure during such events is unacceptable from an economic standpoint. A detailed safety analysis may thus find that the margins provided in the HWR are sufficient to meet U.S. requirements with respect to fuel damage during anticipated and infrequent events, but this is difficult to ensure a priori because it has not been evaluated. It is also appropriate to note that fuel damage is of lesser significance in the HWR because failed fuel can be quickly removed using the on-power refueling feature.

Question 21

Section 2.3.6.1 Operation with the CANDU fuel design, except for higher enrichment with higher burnup, is suggested to lead to higher rates of fuel failure than the Canadians have experienced in the past. What test data are available on failure rates at high burnup? What steps will be taken in design, fabrication, and operation to keep this failure rate acceptable? What level of contamination of the primary coolant system is expected from various failure rates and how is this controlled?

As you know, the currently used General Design Criteria rule out designs that include fuel failure as a normal occurrence. Reconsideration of this position would be expedited by any information you might develop regarding the effect of routine fuel failures on subsequent accident consequences, such as might occur by way of contamination of primary coolant. Even if this criterion were reconsidered, it would seem reasonable to require that a predicted failure rate be low enough so that one damaged fuel element could be expected to be removed before the problem was compounded by additional failures.

Response

Section 2.3.6.1 was not meant to imply that significantly higher fuel failure rates are anticipated as a result of increased burnup; rather, this was intended to be simply a statement of the obvious fact that longer irradiations offer greater opportunities for fuel failure. Unpublished Canadian test data indicate that CANDU fuel can accommodate the higher burnups of the slightly enriched case without compromising fuel performance; however, the number of rods irradiated to such burnups are relatively low and additional irradiations would have to be performed before a statistically valid conclusion could be reached. Evaluations performed to date indicate that no design changes to CANDU fuel would be required for operation to burnups of about 20,000 MWd/MT (Section 4.8 of Ref. 1); further test irradiations should be performed to confirm this conclusion as part of an R&D program.

In summary, fuel will be designed to rule out failure as a normal occurrence. This would be necessitated by economic considerations even if general design criteria were reconsidered. It is also appropriate to recognize that the level of contamination in the primary coolant system can be maintained at rather low levels even in the event of fuel failure (as an abnormal occurrence) because of the ability to replace failed fuel while the reactor remains at power. This is in contrast to the LWR where removal of such failed fuel necessitates a time-consuming and costly refueling outage in which the reactor must be shut down and the head and upper guide structure removed to replace fuel. Under Canadian licensing, primary coolant contamination from fission products is limited by the site boundary doses which would be obtained in a LOCA event assuming containment impairment; in practice, primary coolant activities have been a small fraction of this allowable limit. Other accidents which are dependent on the level of contamination of the primary coolant typically involve steam generator tube leakage. Because of the cost of heavy water, significant steam generator tube leakage cannot be tolerated in the HWR, and so the transport of primary system contamination to the steam side of the plant is unlikely to be significant.

Fuel will be designed to prevent fuel failure as a normal occurrence; higher failure rates than those observed in the LWR are not anticipated; levels of primary system contamination in the event of fuel failures can be kept lower than those of the LWR by removing failed fuel using the on-power refueling capability of the HWR; and the greater leak tightness of heavy water systems as compared to the LWR should serve to decrease the likelihood of transportation of primary system contamination to the secondary side where it might potentially be released during various accidents.

Question 22

Why is the volume of housekeeping-type low-level waste expected to be so much smaller for the HWR than the LWR?

Response

Lower housekeeping wastes are expected for several reasons:

1. The HWR uses on-line refueling (except for the initial core) for reactivity control. This contrasts to the PWR which uses soluble boron for this purpose. In the PWR, the soluble boron concentration is adjusted throughout the cycle by a bleed-and-feed operation. The coolant bled from the reactor must be processed through demineralizer resins and evaporated to remove the dissolved boron. This produces additional quantities of spent demineralizer resins and evaporator concentrates which require disposal.
2. Failed fuel can be quickly removed from the reactor using on-line refueling in the HWR. This reduces the duty on the demineralizer resins.
3. Because of the cost of heavy water, the HWR is designed to minimize primary coolant (and moderator) leakage, and systems are provided to recover and recycle unavoidable leakages.

Question 23

We are not prepared at this stage to agree or disagree with the statement ". . . in recognition of the fact that the CANDU reactor is considered to be at least as safe as the LWR . . . ."

Response

See response to Question No. 13.

Question 24

Should not the monitoring and control of hydrogen be regarded as a subject to be included with Safety System Research? If not, please expand Section 1.2.8.8 to provide details of description and capacity of equipment and sensors.

Response

Like the LWR, hydrogen gas can be produced as a result of the LOCA, and must be prevented from reaching combustible concentrations in containment. Considerations of hydrogen monitoring and control appear identical to the LWR, and any safety systems research would appear equally applicable to both reactor types. It appears that identical equipment and sensors as prove acceptable for the LWR could be employed in the HWR plant.

Question 25

Section 1.1. (p. 1-4). Is there diversity in the in-core sensors for the "two diverse reactor shutdown systems"?

Response

Diversity in in-core sensors is provided for the two diverse reactor shutdown systems. One shutdown system employs vertically mounted detectors entering the calandria from the top while the second shutdown system employs flux detectors which enter the core horizontally. Vanadium and platinum detectors are currently employed in CANDU units.

Question 26

The potential for a small LOCA event due to on-line refueling malfunction, particularly resulting from a seismic event, should be addressed.

Response

It will be necessary to postulate a small break LOCA event due to on-line refueling malfunction, such as may result from the seismic event. A design of the refueling machine, however, is such as to preclude mechanical movements which could simultaneously fail neighboring fuel channels. In this instance, coolant discharge is limited to the diameter of the feeder pipes that connect each pressure tube to the inlet and outlet headers; these feeder pipes range in diameter from 1.5 to 3.354 inches. Thus, a LOCA event due to on-line refueling malfunction is similar to any small break which must be accommodated in reactor design.

PART II

RESPONSES TO SPECIFIC BROOKHAVEN NATIONAL LABORATORY REVIEW COMMENTS

1. Question on natural circulation

In the event that all forced shutdown cooling capability is lost in the HWR, it is claimed that heat removal from the primary system via natural circulation will suffice (p. 59 and p. 266 of the preliminary design report). Although the steam generators are positioned above the core, the fact that the pressure tubes are horizontal raises some obvious questions. Thermal buoyancy effects originating within the tubes will be in a direction orthogonal to the desired direction of the flow. Any steam generated within the tubes during a transient will tend to flow upward toward the top of the tubes and may stagnate there due to the lack of sufficient flow to overcome two-phase frictional resistance. On the other hand, in a pressurized LWR with a vertical core arrangement, the thermal buoyancy of the steam will tend to promote its removal from the core region. One must therefore conclude that with respect to this circumstance, the potential for dryout is greater in the HWR than it is in the PWR. Further remarks on the behavior of bubbles in the primary system are discussed below.

It would, of course, be of interest to learn of Canadian experience with respect to natural circulation in the CANDU reactors. We spoke to C-E about this and apparently the Canadians claim that the CANDU reactor has a natural circulation capability. However, we were not able to receive information on documentation that would substantiate this apparent claim. Fred Jesick (of C-E) also noted to us that the NASAP HWR design is not sufficiently detailed to assess its potential for natural circulation decay heat removal capability.

Response

The first part of this question has been discussed previously. The issues raised in the latter part of this question require some response.

The conceptual and standard HWR designs incorporate power-operated relief valves located between the outlet headers and the pressurizer isolation valves (see Figure B-1). Because of the presence of these power-operated relief valves, the intact coolant loop cannot be overpressurized as a result of loop isolation during a LOCA event.

It is true that if a loss-of-flow event occurs in one primary loop, the potential for providing forced circulation via the other primary loop is small because the loops are interconnected only by the surge lines to the pressurizer. However, the system is designed (as are LWRs) to accommodate the complete loss-of-flow event (i.e. loss of all primary coolant pumps), and so the loss-of-flow event in one loop is just a subset of this more demanding case. More frequent events are expected to be partial loss of flows resulting from the loss of one coolant pump. Since each coolant loop contains two pumps, the loss of one coolant pump would still provide forced circulation in both primary loops.

2. Question on primary heat transport system

The primary heat transport system is a two-loop design with two pumps, two steam generators, two inlet headers and two outlet headers on each loop. According to Fig. 5.1.3-3 of the preliminary design report and our conversation with Fred Jesick, the two loops are connected at a location common to two outlet headers (one from each loop) and the coolant system pressure is controlled by one pressurizer which is also common to the two coolant loops at this location.

If a LOCA event occurs on one loop, then it is possible, via valves provided at the common location to isolate the damaged loop from the intact loop such that the pressurizer is connected only to the intact loop (or isolated entirely) and the damaged loop is then valved to ECCS operation. Isolation of the pressurizer from the intact loop would affect system pressure control in that loop and, therefore, would not be recommended by us.

If a loss-of-heat-sink event occurred in the secondary coolant system such that the initially intact primary heat transport system became effectively adiabatic and system temperature and pressure began to rise (in both loops), then it is expected that the pressurizer relief valve would open and the pressure would be relieved. If this relief valve failed to re-close after the primary system pressure was reduced to a safe level, then the accident becomes a loss-of-coolant accident via the opened pressurizer valve. An obviously important distinction between the course of events for this hypothetical accident and the accident that occurred recently for a pressurized LWR is that the loss of coolant in the HWR is associated with a positive void reactivity feedback coefficient.

A loss-of-coolant accident at the pressurizer is particularly important since the complete loss of coolant from both loops (but not from a single loop) results in a reactivity insertion greater than one dollar.

Because of the presence of the loop isolation valves, certain variations of the above scenario become possible. For example, if, due to the observation of this loss-of-coolant accident, it is decided that (for whatever reason) one of the loops should be isolated from the pressurizer and the other loop, then, due to the continued presence of the loss-of-heat-sink condition, the isolated loop could overpressurize and be breached in a manner that would compromise coolability via that loop.

If a loss-of-flow event occurs in one primary loop, it should be noted that, since the loops are in common only at two outlet headers, the potential for providing forced circulation via the other primary loop appears to be small.

Response

See response to Part I, Question 2, and Part III

3. Question on moderator cooling.

The moderator cooling system of the HWR provides a heat sink which is not available in the pressurized LWR. Even if flow is not available in the moderator system, it may function as a passive heat sink following a loss-of-heat-sink accident as described above. However, the efficacy of the moderator system as a heat-removal path under a spectrum of conditions cannot be evaluated by us at this time due to a lack of sufficient design information. A comparison of the HWR and PWR in this regard should recognize the existence of upper and lower plena in the PWR as additional heat sinks not available in the HWR. The process of uncovering the core via steam production is quite different for the two designs and the analysis of available heat sinks must be analyzed with care.

Response

This question was also discussed previously, but some emphasis should be made that the use of the moderator as a heat sink is appropriate only in the context of ECCS failures, so as to avoid what would otherwise be a Class 9 event. Heat transfer to the moderator is not a viable alternative to a normal heat sink such as the steam generators or shutdown cooling system; it functions only when the temperatures of the fuel and pressure tube have reached sufficiently high values that pressure tube slumping and contact with the calandria tube occurs.

4. Question on a loss-of-heat-sink scenario.

By considering failure in the secondary coolant system similar to that which occurred at the Three Mile Island plant on March 28, 1979, the possible situation that may exist in the primary loop of the NASAP HWR is discussed as follows. The discussion is based on the information included in Chapter 5 of the HWR preliminary design report.

If the primary loops are overheated and intensive boiling causes bubbles to form in the pressure tubes, these bubbles cannot be removed from the core (pressure tube) as easily as in the PWR system where bubbles are carried upward by thermal buoyancy. The bubbles would either stay in or flow through pressure tubes which may aggravate the heat transfer from the cladding and enhance the temperature increase. Because of the structure and the layout of the inlet and the outlet headers, it is not likely that large bubbles would be formed there. Bubbles entering the outlet header/inlet header would be expected to enter into the loop/pressure tubes through the hot leg/cold leg of the steam generator. It is also not expected that bubbles downstream of the outlet headers would enter the pressurizer (and be released) any easier than in the PWR system.

It is believed that, in the HWR, any intensive bubbles that are formed would mostly circulate along the loops through headers, steam generators, main pumps, and pressure tubes. This may not only enhance the temperature increase in pressure tubes but also cause pumps to cavitate. Without forced circulation, the potential for natural circulation could be reduced or even could be blocked by the existence of a large number of bubbles. The reduced flow would cause more overheating and/or damage of fuel elements and pressure tubes.

As discussed above, the moderator inventory is an alternate heat sink in the HWR system. However, this heat sink is not in direct contact with the coolant and the pressure tubes (there is a He-gas-filled space between the pressure tube and the guide tube) and thus may not provide a sink which would respond quickly enough to preclude bubble formation.

If a meltdown occurs, then the potential interaction of the moderator system with the core debris would require investigation.

Response

See response to Part I, Question 4, and Part III.

5. Question on common mode heat removal failure at headers

In the HWR, there is a low-pressure injection system (LPIS) and there is a high-pressure injection system (HPIS) to protect the core during a LOCA event. These systems provide borated water to both inlet and outlet headers. There are hundreds of tubes connecting each header to the pressure tubes via welds. Based on our limited information on the design, we note that if a LOCA event is initiated by a failure of an inlet header, then it is possible that this failure may also prevent enough emergency cooling water from entering the cooling channels connected to the failed inlet header. The potential for this common source for losing cooling ability should be investigated further.

Response

This item has been previously discussed in Part I, Question 5. The simultaneous injection of emergency core cooling water into both the inlet and outlet headers ensures that emergency cooling water must pass through the cooling channels before exiting through a failed header.

6. Question on temperature and void coefficients

The (preliminary) temperature and void coefficients computed at BNL for the C-E design 1.2 wt% U-235 PHW fuel bundles are shown in the table below (in units of $10^{-5}\Delta\rho$).

Table B-1. Temperature and void coefficients

	Burnup Cycle			
	Start	Middle	End	Reactor Average
Fuel (Doppler) coefficient (per ^{o}C)	-1.0	-0.6	+0.2	-1.4
Coolant temperature coefficient (per ^{o}C)	+2.0	+3.5	+5.7	3.7
Moderator temperature coefficient (per ^{o}C)	-5.8	+0.14	+11.0	+1.8
Coolant void coefficient (100% void)	+1,100.0	+850.0	+850.0	~850.0

The fuel (Doppler) coefficient is negative at the start and middle of the cycle and slightly positive at the end of the cycle. The reactor average value of the Doppler coefficient (averaging all fresh and high burnup bundles in the reactor) is negative. The coolant temperature coefficient is positive at all times in the burnup cycle. The moderator temperature coefficient is negative for fresh bundles but positive for bundles that have achieved more than half their design burnup. In the equilibrium cycle of continuous refueling, the moderator temperature coefficient is positive. The at-power coolant void coefficient (at 100% void) represents a reactivity of ~\$1.30.

The large mean neutron generation time ~10^{-3} sec in the PHWR mitigates the effect of the positive temperature coefficients by providing time for the control or safety systems to respond to small changes in temperature.

The following (preliminary) table illustrates the effect of coolant or moderator temperature increase.

Table B-2. Effect of temperature increase on reactivity, prompt power, and reactor stable period

Coolant Temp. Increase, °C	$\Delta\rho$	Prompt Power Increase, %	Reactor Stable Period, sec
1	3.7	0.6	2,250
10	37	6.0	214
50	185	40	32
Moderator Temp. Increase, °C			
1	1.8	0.3	4,650
10	18	2.8	454
5C	90	16.1	80.4

The loss of moderator cooling will have a positive reactivity effect, but there appears to be sufficient time to sense the moderator temperature change and shut down the reactor.

Temperature increases in the coolant would be accompanied by temperature increases in the fuel, resulting in a reactivity increase of about half that shown above for the coolant temperature increase. For slow increases in coolant temperature, there appears to be sufficient time to control the power.

In a LOCA event where both coolant loops are voided, the PHWR will be prompt critical with a reactor period ~0.5 seconds. In one second the power would increase by a factor of 10. If only one loop lost coolant, the reactivity insertion would be approximately $0.65. This would cause a 65% increase in power within one second of voiding one coolant leg and increasing the power by about a factor of 10 within 7 seconds unless the reactivity transient is stopped by the safety systems. There appears to be entirely too much reactivity associated with voiding one loop of the two-loop design PHW. As a comparison, voiding a single loop in a four-loop system would double the power in about 7 seconds, providing more time for safety systems to respond.

Response

These items were also discussed previously in Part I, Question 7. The C-E and BNL calculations are in essential agreement. The response of the HWR to the LOCA event is determined only partially by the void coefficients; it is also determined by the rate of blowdown which establishes the time-dependence of void formation and hence of reactivity insertion through the positive void coefficient. Because of this time-dependency of void formation, the transient can be terminated by reactor trip before significant increases in core power level have occurred.

7. Question on production/discharge data (preliminary).

The following table compares the annual discharge of HWR fuel and LWR fuel, based on thermal power of 4,029 MW. Although the data given here is preliminary, the estimates are approximately within 10% of the C-E values.

Table B-3. Annual discharge data for HWR and LWR fuel

	HWR	LWR
Burnup MWd/MT	19,750	29,789
Total discharge (kg)	56,516	37,025
U-235 (kg)	74.3	328.2
U-236 (kg)	87.3	129.3
U-238 (kg)	54,920	35,144
Pu-239 (kg)	170.1	197.5
Pu-240 (kg)	115.2	76.1
Pu-241 (kg)	29.6	42.5
Pu-242 (kg)	21.5	15.8
Total Pu (kg)	336.5	331.9

The HWR discharges 1-1/2 times more burned fuel by weight than the LWR, thereby increasing the volume of waste to handle. The total amount of plutonium produced is about the same in the HWR as the LWR. In the HWR, 59% of the discharge Pu is fissile, while 72% of the discharge Pu of the LWR is fissile material. The relatively larger amounts of Pu-240 and Pu-242 in the HWR fuel make it less suitable for recycle or weapons purposes than LWR discharge fuel.

Response

The C-E evaluations are in essential agreement with the BNL calculations.

8. Question on xenon oscillations.

In this section, the control problem associated with the xenon instability in the C-E HWR is summarized. More detailed information can be provided if necessary.

The neutronic dimension of the CANDU reactor is about 4 times larger than the pressurized LWR and the oscillation of power distribution, due to xenon concentration build-up and decay, becomes a serious problem for reactor operation. In the C-E HWR, the total electric output is 1,250 MWe which is about twice the output of the current CANDU reactor. Therefore, the physical size of the core would be twice the size of the CANDU reactor. Furthermore, the enrichment of U-235 in the fuel rods is 1.2% instead of natural uranium in CANDU fuel. This results in a migration area about 6% smaller than in the CANDU reactor. Thus, the neutronic dimension of the C-E HWR is more than twice that of the CANDU reactor and thus the higher harmonic xenon oscillations will be excited. In order to control these xenon oscillations, a control mechanism such as the water compartment used in the CANDU reactor should be used. The number of control zones (water compartments) will be increased from 14 in the current CANDU to 32 in the C-E HWR. As the number of control zones increases, the self-powered in-core detectors such as vanadium and platinum detectors will be increased from 100 to 28, respectively, to 230 and 64, respectively. The size of the computer that controls the flux distribution should be increased in approximate proportion to the neutronic size of this reactor.

Response

Combustion Engineering, Inc., is in essential agreement with the BNL assessment. Although the use of slightly enriched fueling will tend to reduce the problem associated with xenon instability (because of the lower thermal flux), control of xenon instabilities is expected to be more of a problem in the C-E HWR design because of its larger size. It is for this reason that the number of control zones and in-core detectors have been increased, and this will undoubtedly result in the need for increasing the size of the control computer.

9. Question on neutron behavior associated with the loss-of-coolant accident.

The current LWRs have vertical coolant channels but in the CANDU type HWR, the fuel rods are oriented horizontally. In addition to the limited heat transfer data (available in the open literature) for rods having horizontal flow, the flow patterns in horizontal tubes are significantly different from the vertical flow patterns.

The void coefficient of the reactivity change is a very important quantity to analyze for the neutronic behavior in the case of a loss-of-coolant accident. The stratification of voids inside tubes will affect the neutron transport inside the core. Furthermore, the neutron streaming effect, due to void stratification, will change the void coefficient which usually is calculated under the assumption of homogeneous void.

Response

See response to Question 9, Part I.

PART III

RESPONSES TO BROOKHAVEN NATIONAL LABORATORY DISCUSSED DIRECTLY WITH COMBUSTION ENGINEERING, INC.

With respect to the overall design and functionability of the shutdown heat removal system, this following information is needed:

- The heat removal capability of the system if less than four steam generators are operational
- Whether the main heat transport system can be used during cold shutdown if the shutdown cooling system (the analog of the residual heat removal system in the PWR) fails
- Whether an adequate heat sink can be provided by the condensate pumps and the safety valves in connection with the steam generator if the main feedwater pumps fail. Number of safety valves (out of five) required to open. Number of safety valves together with atmospheric relief valves required to open if all other heat sinks are not available upon failure of both the main feedwater pumps and the condensate pumps.
- How the electrical system (both AC and DC) is connected to the various safety loads and control systems.

1. Question on primary loop natural circulation capability--(bases for capability; is there enough?)

Response

A detailed evaluation of the available natural circulation heat removal capabilities of the HWR has not been undertaken. A rough estimation of the available capacity for heat removal by natural circulation, however, indicates that adequate heat removal would be available to remove the decay heat that would exist following RCP coastdown, assuming the reactor tripped from 100 percent power at the same time that all four pumps lost power. This estimation is conservative in that subcooled heavy water was assumed when in fact saturated water could exist.

2. Question on the capability of heat removal with loss of heat sink from three steam generators

Response

The HWR coolant system contains two separate coolant loops each containing two steam generators, two pumps, and one-half of the reactor fuel channels and end connections. The only connection between these two loops is the surge line. Therefore, if under any condition three of the four steam generators become inoperable, cooling would be available for only the loop containing the remaining steam generator.[a]

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[a]This neglects moderator cooling of fuel.

To ensure that each loop will have cooling by a minimum of one steam generator, each steam generator will be provided with two redundant atmospheric dump valves. Each dump valve will be capable of removing decay heat and RCP heat, and cooling one loop down to shutdown cooling initiation temperature and pressure. No single active failure will prevent operation of at least one dump valve per steam generator. Main steam safety valves will dissipate the system energy if the steam pressure reaches their set pressure prior to opening the dump valves. Make-up water to the steam generator is ensured by the use of redundant emergency feedwater supplies which meet single failure criteria.

3. Question on use of main cooling system at low temperature and pressure 30 minutes after failure (assumed above shutdown cooling initiation pressure and temperature)

Response

There will be no restrictions on RCP operation at any temperature above 70°F except adequate pump suction pressure such that the required pressure across the pump seals and/or net positive suction head requirements are met. Following a primary pipe break, operation of the pumps would depend upon the capability to maintain the loop pressure discussed above. The intact loop would be unaffected. Furthermore, it should be noted that emergency power is not provided for the RCPs.

4. Question on the reliability of pressurizer isolation values against improper activation

Response

The four isolation valves in the surge line, two per loop connection, are provided to ensure that the unaffected loop and pressurizer do not blowdown following a pipe rupture in one loop. These valves are conceived as being air-operated with a safety-grade air source and fail closed upon loss of air or electrical power. Each valve in the affected loop will receive a different signal to close in the event that a loss-of-coolant accident occurs. This signal will be based on low reactor coolant header pressure in conjunction with high containment pressure. The exact signal generation basis will depend upon detailed accident analyses which have not been conducted. Emergency power and safety grade air sources will ensure that the valves in the intact loop will remain open or be reopened, by the operator to provide pressure control during cooldown of the intact loop. The closure time of these valves will be specified as a maximum of 5 seconds. Overpressure protection for the reactor coolant system (RCS) is provided by relief valves on the reactor side of these valves.

5. Question on ensuring flow of coolant by flywheels on reactor coolant pumps during loss of offsite power

Response

Each RCP will be supplied with a flywheel to increase the rotating inertia of the RCP assembly. This inertia increases the reactor coolant pump coastdown time and reduces the rate of reactor coolant flow decay if electrical power to the RCP motors is lost.

6. Question on safety-related electrical system equipment

Response

Figure 5.1.6-1 (Ref. 1) includes a dark dashed boundary line to identify all equipment which is nuclear safety related.

7. Question on how many of the five safety valves in the main feedwater pumps are required to open

Response

This issue cannot be addressed at this time because a transient analysis has not been accomplished for the HWR concept.

The conceptual design provided main steam safety valves with the capacity expected, in conjunction with the primary relief valves, to provide overpressure protection for the steam generator and the RCS following a complete loss of turbine/generator load. This assumes that the non-safety-related pressure relief systems do not operate and a reactor trip on high pressure occurs. This design primary and secondary relief capacity has been found to be adequate on pressurized LWR plants to provide overpressure protection for a total loss of feedwater transient.

In either case, the atmosphere dump valves (relief valves) would be used to cool down the steam generators and the reactor coolant system. These valves, installed to facilitate plant cooldown, will not be sized to provide overpressure protection.

8. Question on how many safety and relief valves are needed to operate in conjunction with emergency feedwater pumps

Response

The secondary or main steam safety valves are designed to provide overpressure protection and thus energy removal in the interval between the initiation of the transient and the shutdown of the reactor. Adequate capacity will be provided for the worst case transient--loss of load with the control systems not working. The emergency feedwater system may or may not be operating during this portion of the transient. The exact number of safety valves will depend on the given transient which has not been analyzed as yet.

Following reactor shutdown and isolation of the steam generators, the atmospheric dump valves (relief valves) in conjunction with the emergency feedwater pumps function to remove decay heat and to cool the steam generators and the reactor coolant system to shutdown cooling system initiation conditions.

REFERENCES FOR APPENDIX B

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