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LICENSING EVALUATION OF CANDU-PHW NUCLEAR POWER PLANTS RELATIVE TO U.S. REGULATORY REQUIREMENTS*

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ABSTRACT

Differences between the U.S. and Canadian approach to safety and licensing are discussed. U.S. regulatory requirements are evaluated as regards their applicability to CANDU-PHW reactors; vice-versa the CANDU-PHW reactor is evaluated with respect to current Regulatory Requirements and Guides.

A number of design modifications are proposed to be incorporated into the CANDU-PHW reactor in order to facilitate its introduction into the U.S. These modifications are proposed solely for the purpose of maintaining consistency within the current U.S. regulatory system and not out of a need to improve the safety of current-design CANDU-PHW nuclear power plants.

A number of issues are identified which still require resolution. Most of these issues are concerned with design areas not (yet) covered by the ASME code.

Introduction

The Canadian approach to nuclear reactor safety, while having been influenced by developments elsewhere has, to a large extent, been developed independently.[1-27] This fact, combined with the fact that the CANDU-PHW (CANDU-Pressurized Heavy Water) reactor has a number of intrinsic safety-related characteristics differing substantially from those of other commercial nuclear power reactor types, has led to the development of Canadian licensing criteria and safety design bases which are in some areas different from those developed, e.g., in the U.S. for light water reactors (LWRs).

CANDU-PHW power plants are, as part of the Canadian licensing process, submitted to a rigorous safety review performed by the Atomic Energy Control Board (AECB), the nuclear regulatory body of Canada. The primary issue in evaluating CANDU licensability in the U.S. is therefore not whether CANDU reactors meet adequate safety standards, but rather how much effort may be required to introduce the CANDU technology into the U.S. regulatory environment, as determined by current U.S. NRC regulatory criteria, guides, standards, procedures, and practices.

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The evaluation of the licensability of U.S.-sited CANDU reactors may be approached in a number of ways. One way would be to follow a purely probabilistic approach of the WASH-1400 type, comparing over the entire spectrum of postulated accident sequences the overall probability of each type of accident sequence for CANDU reactors and LWRs. This approach, which would appear to have considerable merit, has certain difficulties associated with its application. One important difficulty is that current U.S. licensing procedures are up to now only to a relatively small degree based on probabilistic considerations. The approach followed in this paper, which seems to parallel to some extent that applied up to now for the Liquid Metal Cooled Fast Breeder Reactor (LMFBR) in the U.S., is that of equivalency of safety. This is, for the purpose of this paper, understood to mean that a U.S.-sited CANDU reactor is to have, in all areas, safety levels (margins) that are equal to, or higher than, those of U.S.-sited LWRs, making proper allowances for the differences between the safety-related intrinsic characteristics of CANDU reactors and LWRs.

It should be recognized that U.S. Regulatory Design Criteria and Guides (including the General Design Criteria, Title 10 of the Code of Federal Regulations, Part 50, Appendix A) have, to a very large extent, been developed for the present generation of LWRs. In view of this, some of these criteria and guides are not (or only partially) applicable to CANDU reactors, while others should be interpreted as to their intent rather than their specific wording.

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

Brief Description of CANDU-PHW Reactors

CANDU-PHW reactors belong to the family of heavy-water-moderated pressure-tube reactors. Other members of this family are: CANDU-BLW (CANDU-Boiling Light Water), CANDU-OCR (CANDU-Organic Cooled Reactor), SGHWR (Steam Generating Heavy Water Reactor), CIRENE (CISE REattore a NEbbia), EL-4 (Eau Lourde-4), FUGEN, and ORGEL (ORGanique-Eau Lourde, discontinued program).

Among the main characteristics of the CANDU-PHW should be named (1) use of pressure tubes in the core region, (2) use of D_2O as moderator (cold, unpressurized), (3) use of pressurized ($\sim$ 1400 psia) D_2O as coolant, (4) use of natural uranium as fuel (could be slightly enriched - $\sim$ 1.2% - if desired for higher burnup -

~ 20,000 MWd/ton), and (5) use of an on-load refueling scheme.

Figure 1 gives a simplified flow diagram showing the main characteristics of the Primary Heat Transport System (PHTS). The PHTS may consist of one, two, or more independent subsystems (loops); Figure 1 shows the case of two loops. It is noted that each loop consists of a "figure-eight configuration comprising as main components: Two pumps, two steam generators, four headers, and a large number of feeder lines and power channels. The CANDU-PHW reactor has as one of its attractive features the capability of permitting, during normal operation, location and removal of failed fuel elements that have become a source of radioactive contamination of the PHTS.

Figure 2 shows the level of seismic qualification and the degree of separation of the various subsystems. It is noted that the standard CANDU-PHW nuclear power plant is provided with two sets of emergency diesel generators [denoted: (1) on-site standby diesel generators, and (2) emergency power supply - EPS] with different seismic qualifications. Similarly, two separate sources for emergency water supply are provided, one of which is seismically fully qualified. For a more detailed description of the CANDU reactor, reference is made to the literature on this subject. CANDU-PHW reactors are provided with two independent and diverse shutdown systems (SDS-1 and SDS-2).

Evaluation

In evaluating the safety aspects of CANDU reactors with respect to licensability in the U.S., it is desirable to clearly distinguish design characteristics that are intrinsic (Table I) to the Nuclear steam Supply System (NSSS) from those that are non-intrinsic (i.e., pertaining to subsystems of a more peripheral nature), and that could therefore be modified relatively easily without changing any of the principal characteristics of the NSSS.[27] In the latter category are the design characteristics associated with such systems as containment, most parts of the control and plant protection, auxiliary feed water supply, most engineered safeguards, etc.

The principal differences between the Canadian and U.S. safety approach pertain to the following areas:

(1) *Safety Analysis:* The Canadian licensing procedures require analysis of dual-failure accidents (i.e., failure of a process system coincident with failure of a safety system), including the occurrence of a maximum-size LOCA simultaneously with unavailability of the Emergency Coolant Injection System (ECIS), or impairment of the containment system (Table II). A further difference between Canadian and U.S. licensing

criteria relative to safety analysis pertains to the maximum permissible conditions for the cladding attained during or following blowdown in case of a postulated LOCA. One of the U.S. requirements is that the cladding temperature shall not exceed 1200°C at any point in the core and at any time during or following the blowdown, whereas the Canadian requirement (bearing on oxygen-embrittlement of the cladding)[22-24] is more flexible, having the character of a time-at-temperature limit rather than a strict temperature limit. This difference appears, however, to be not essential because Canadian LOCA analyses, performed for current-design CANDU-PHW reactors, show the maximum cladding temperatures to be about 1200°C. Furthermore, the underlying intent of the U.S. ECCS criteria is maintaining coolable core configuration, which for CANDU-PHW reactors can be accomplished, even in case of unavailability of the ECIS, by the Moderator Cooling System.

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

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

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

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

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

A number of issues remain to be resolved if CANDU-PHW reactors are to be introduced into the U.S., including:

(1) Compatibility of the CANDU-PHW reactor's design with the current ASME B&PV Code; examples of items requiring resolution are: (a) use of Zr-Nb alloy in the PHTS (for the pressure tubes), (b) use of rolled joints in the PHTS, (c) use of special stainless steels in the PHTS (e.g., use of special 403 SS for the endfittings of the pressure tubes), (d) use of Zircaloy-2 for calandria tubes, control rods guides, and injection nozzles of the second shutdown system (gadolinium injection), (e) required ASME-Section III-Class (1 or 2) for the calandria;

(2) required degree of in-service inspection for the feeders and the end-shields of the calandria.

Most of the unresolved issues appear to be related to design solutions applied in the CANDU design which are not yet covered by the ASME code. The reason for this is historical: The nuclear sections of the ASME code have, to a large extent, been developed in response to the needs of the LWRs; there was, up to now, no great need in the U.S. for code development in areas of interest to pressure tube reactors. Canada has developed its own codes and standards[21] relative to design aspects not covered by the ASME code.

Conclusion

CANDU-PHW reactors have a number of intrinsic safety features which should facilitate its possible introduction into the U.S. Among these features should be named the availability of a large heat sink, which is intimately dispersed in the core region (the moderator), and which is provided with a cooling system that could serve as backup in case of failure of the primary emergency core cooling system, denoted ECIS in CANDU-PHW reactors.

In case of siting in the U.S., CANDU-PHW nuclear power plants may have to undergo some design modifications, affecting primarily some of the peripheral subsystems, and none of the intrinsic characteristics. The motivation for such design changes follows primarily from a need to maintain consistency within the U.S. regulatory system (particularly as regards requirements for seismic qualification), and does not reflect a need to improve the level of safety of current-design CANDU-PHW nuclear power plants.

It is concluded that CANDU-PHW nuclear power plants can be introduced into the U.S.; the time scale on which this can be accomplished appears to depend not so much on the technical issues involved, but more so on whether there exists a sufficient economic incentive to do so.

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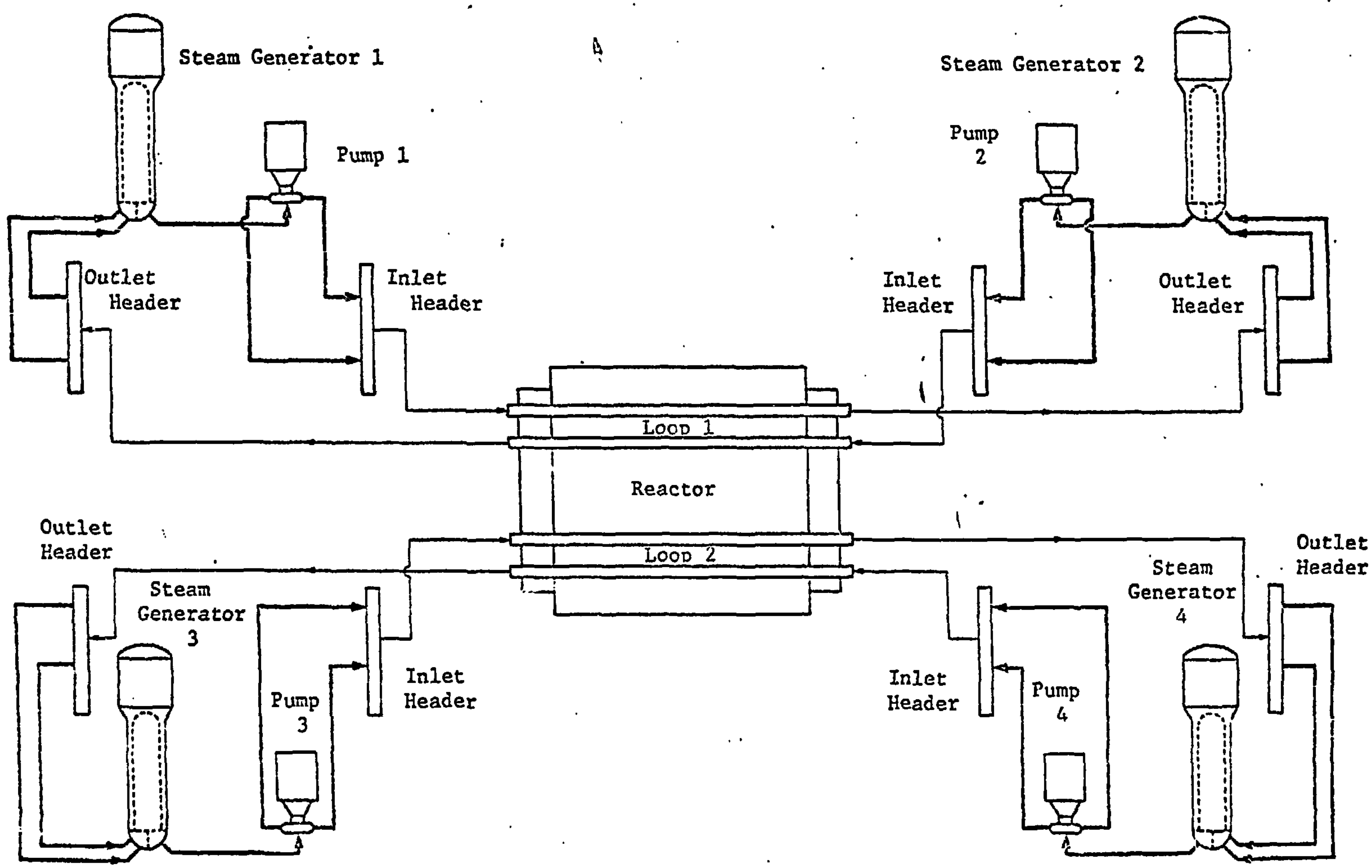

Fig. 1. Primary Heat Transport System - Main Circuit Flowsheet

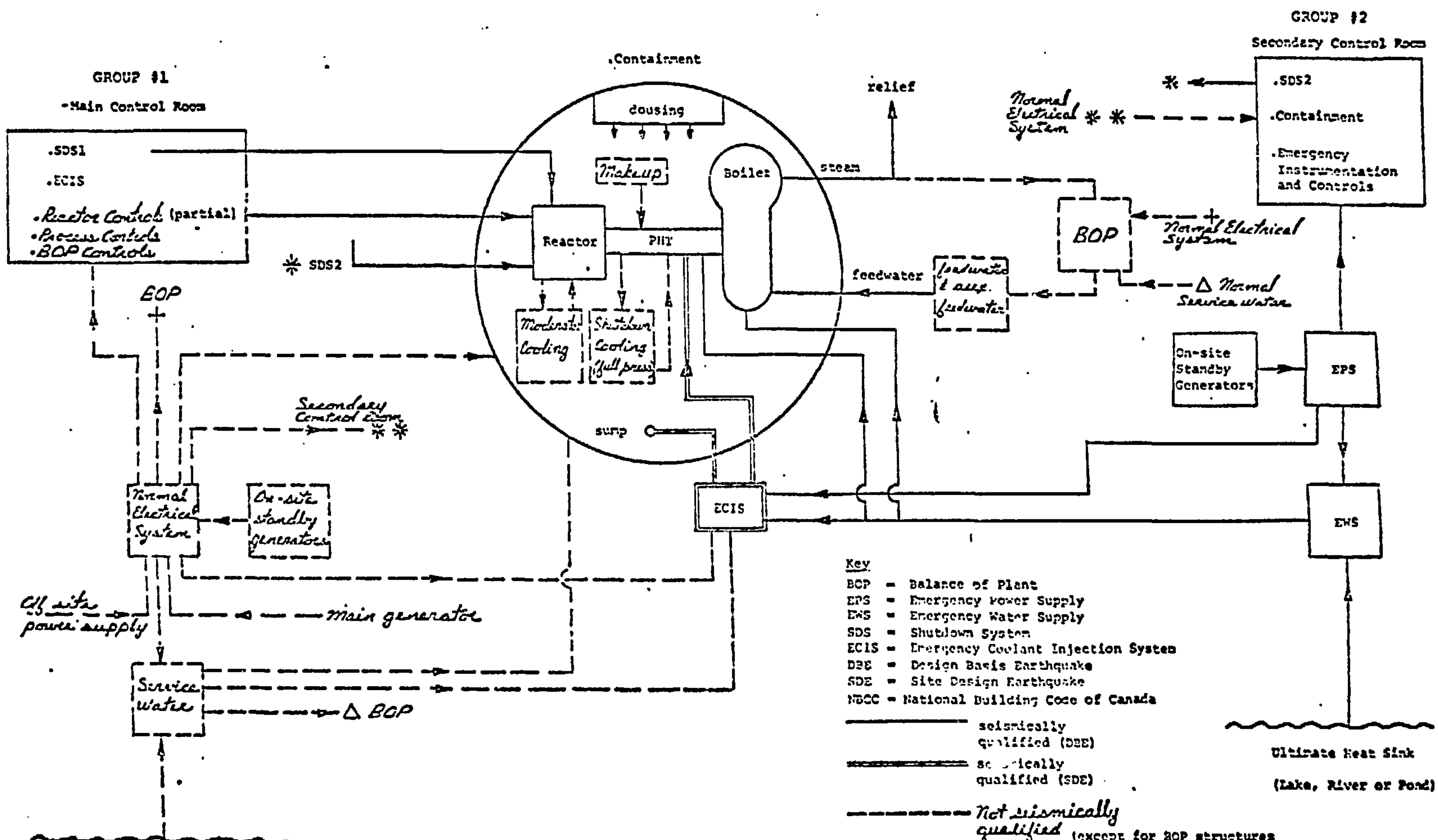

Fig. 2. Seismic Qualifications and Systems Separation for a 600-MWe Station on an Ocean Site

Table I. Some Important Safety-Related Intrinsic Characteristics of CANDU-PHW Reactors

	Characteristics	Safety Implications
1.	The pressure tubes (which are part of the primary coolant pressure boundary), traverse the active core region.	Stress-bearing components of the coolant pressure boundary are subjected to the full neutron flux. Failure of a pressure tube in the core region may have to be considered as part of the safety evaluation. The probability for tube-to-tube failure propagation in the core region must be very low.
2.	The pressure tubes (having a relatively small wall thickness) have leak-before-break characteristic.	The probability of a sudden large-size break in a pressure tube is very small, because the tube will first develop a leak.
3.	The pressure tubes are surrounded by calandria tubes, creating a gas-filled annular space between the tubes.	A crack in a pressure tube, resulting in primary coolant leakage, is easily detected by means of the surveillance system analyzing the gas contained between pressure tubes and calandria tubes.
4.	The core is subdivided in separate channels having individual coolant supply.	The primary cooling system can be subdivided into a number of subsystems, thus limiting complete blowdown to only a part of the core in case of a loss-of-coolant accident (LOCA). LOCA is mitigated by hydraulic resistance in piping, in part due to the figure-eight layout of the PHTS. The ECIS is capable of delivering emergency coolant to all core locations with low probability of performance failure. The simple configuration of the power channels (pressure tube + fuel) allows relatively easy testing of ECIS performance (scaling is relatively easy). Failed fuel can be easily detected and located.
5.	Large inventory of cold moderator, having redundant cooling system with capability of removing decay heat.	Large dispersed heat sink in core region. Moderator heat capacity and cooling system serve as a diverse back-up system for ECIS.
6.	Moderator region is surrounded by large light water shield region, having large heat capacity and redundant cooling system.	Provides additional heat sink close to core region, which could serve as back-up system for ECIS and moderator cooling system.
7.	Total excess reactivity is small for natural-uranium-fueled equilibrium core.	Relatively mild power excursions due to accidental reactivity insertions.
8.	Power-reactivity coefficient at nominal power level is close to zero, and may be slightly positive.	Power transients due to uncompensated reactivity insertions would tend to be not self-limiting.
9.	Void-reactivity coefficient of the coolant is positive.	LOCA leads to a reactivity increase. Under-cooling transients lead to reactivity increase due to boiling in power channels.
10.	Mean neutron lifetime is $\sim 10^{-3}$ sec, i.e., $\sim$ 30 times larger than for LWRs.	Power transients tend to be, for the same reactivity insertion, less severe for CANDU reactors than for LWRs.

Table I. Some Important Safety-Related Intrinsic Characteristics of CANDU-PHW Reactors (Contd.)

	Characteristics	Safety Implications
11.	The neutron poison devices for control and safety shutdown are installed in the low-pressure moderator region.	There is no pressure-assisted reactivity accident associated with the control or shutdown rods (compare with rod-ejection accident in LWRs). Control rods and safety shutdown rods are not subjected to hydraulic forces in case of an ex-core LOCA (contrary to what is the case for LWRs).
12.	On-load refueling.	Failed fuel can be easily replaced without necessitating reactor shutdown, thus providing a means for maintaining a low level of radioactivity in the PHTS. Refueling malfunctions could result in small-scale LOCA. Jamming of fuel subassembly during refueling operation could result in undercooling incident, affecting a single channel. Fueling machine becomes part of the PHTS during refueling operation, requiring appropriate seismic design for the site in question, which has been provided. On-load refueling results in a reactor system with relatively low control reactivity requirements.
13.	Burnup of fuel is low (<10,000 MWD/ton).	Fission product inventory is relatively small.
14.	Inventory of tritium relatively large.	Requires special attention (however, the major part of the tritium inventory is in the low-pressure moderator region).

Table II. Matrix of Some Postulated Accidents*

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

*Postulated accidents indicated by X require analysis, whereas those indicated by -- are trivial cases.

Table III. Proposed Design Modifications for U.S.-Sited CANDU-PHW Nuclear Power Plants

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

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

(3) Qualify the ECIS to the SSE level;

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

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

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

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

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