POLICY ISSUE
(Notation Vote)

April 8, 1993

SECY-93-092

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: ISSUES PERTAINING TO THE ADVANCED REACTOR (PRISM, MHTGR, AND PIUS) AND CANDU 3 DESIGNS AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

PURPOSE:

To request Commission guidance for those areas where the staff is proposing to depart from current regulatory requirements in the preapplication review of the advanced reactor and CANDU 3 designs.

BACKGROUND:

The Advanced Reactor Policy Statement (51 FR 24643) and NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," define advanced reactors as those with innovative designs for which licensing requirements will be significantly different from the existing light-water reactor (LWR) requirements. These documents also give guidance for developing new regulatory requirements to support the advanced designs. Staff reviews of these advanced reactor designs should utilize existing regulations to the maximum extent practicable. When new requirements are necessary, the staff should move toward performance standard regulations and away from prescriptive regulations. Each designer is encouraged to propose new criteria and novel approaches for evaluating the design. An objective of early designer-staff interaction should be to develop guidance on licensing criteria for the advanced reactor and CANDU 3 designs and to make a preliminary assessment of the potential of that design to meet those criteria.

NOTE: TO BE MADE PUBLICLY AVAILABLE IN 3 WORKING DAYS FROM THE DATE OF THIS PAPER

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The staff is conducting preapplication reviews of the following four designs:

- General Atomics (GA) 350-MWt modular high-temperature gas-cooled reactor (MHTGR) design sponsored by the U.S. Department of Energy (DOE) Gas-Cooled Reactor Program
- General Electric (GE) 471-MWt power reactor innovative small module (PRISM) reactor design sponsored by the DOE Advanced Liquid-Metal Reactor (ALMR) Program
- Atomic Energy of Canada, Limited, Technologies (AECLT) 1378-MWt Canadian deuterium-uranium (CANDU 3) reactor design
- ASEA Brown Boveri-Combustion Engineering (ABB-CE) 2000-MWt process inherent ultimate safety (PIUS) reactor design

The staff has analyzed policy issues and made recommendations in Enclosure 1. The current designs are summarized in Enclosure 2.

Enclosure 3 lists pertinent Commission papers and reference staff documents for these preapplication designs. Some information in the original documents may have been superseded by more recent preapplicant submittals.

Enclosure 4 serves as a guide to the staff's resolution of comments submitted by the preapplicants and other interested parties, and in Enclosure 5, the staff responds to the Advisory Committee on Reactor Safeguards (ACRS) review of Enclosure 1.

In response to a Commission staff requirements memorandum (SRM) (SECY-91-202, "Departures From Current Regulatory Requirements in Conducting Advanced Reactor Reviews," July 2, 1991), the staff committed to identify those policy and technical issues that require Commission guidance or staff resolution for design certification. The staff committed to do this during the preapplication review and to include situations in which advanced reactor designs deviate significantly from current regulatory requirements.

Policy issues for evolutionary and passive LWRs have been stated in the following Commission papers:

- SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," January 12, 1990
- Draft SECY paper (distributed for comments on February 27, 1992), "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements"
- Draft SECY paper (distributed for comments on June 25, 1992), "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs"

DISCUSSION:

In their submittals, the preapplicants described how their designs complied with the current LWR licensing requirements. Where they did not comply, they gave alternative criteria for evaluating the designs. The staff has conducted a preliminary review of the four preapplication designs using existing LWR regulations and the evolutionary light-water reactor (ELWR) and advanced light-water reactor (ALWR) policy guidance. In this initial review, the staff described 10 issues that require policy direction from the Commission because deviations are proposed from existing regulations. In these issues, either existing regulations do not apply to the design or preapplicants are proposing criteria that are significantly different from the current regulations. These issues, background information on current LWR requirements, preapplicant-proposed approaches, staff considerations, and staff recommendations for Commission approval, appear in Enclosure 1.

The staff developed its recommendations by considering information from the preapplicants, the public, and the ACRS. The staff considered the preapplicant proposals in light of the Commission's policy statements and guidance on severe accidents, advanced reactors, and safety goals in order to develop a single consistent policy recommendation for application to all applicable advanced reactor designs. In some instances, the staff recommends that current regulations continue to be applied to the advanced reactor designs despite preapplicant proposals to do otherwise. In some cases, the staff proposes more conservative alternatives to the preapplicant proposals to account for uncertainties associated with the conceptual design, which should ensure that conclusions made during the preapplication review will serve as a reasonable basis for a finding at design certification that the detailed design is acceptable. It is intended that the safety level standards for these designs will be consistent with Commission guidance at design certification.

Some issues are closely related. Accident evaluation and source term provide a basis for containment performance and emergency planning. Approaches taken for residual heat removal and reactivity control are intended to be consistent with the accident evaluation categories and consequences.

The staff proposes to treat the MHTGR, PRISM, and PIUS designs as advanced reactors in accordance with the policy statement. The CANDU 3 design is considered to be an evolutionary heavy-water design deriving from the larger CANDU reactor designs operating in Canada and elsewhere. Therefore, the staff has concluded that a prototype CANDU 3 is not required for design certification. This position is consistent with staff conclusions in SECY-89-350, "Canadian CANDU 3 Design Certification," November 21, 1989, and SECY-90-133, "Prototype Requirement for CANDU 3 Design," April 6, 1990. The staff has been informed of Canadian plans to build a CANDU 3 plant in the Province of Saskatchewan. However, AECLT has stated that its application for a standard design certification will be independent of any schedule to build a reference plant and the NRC review for such a certification should only need the relevant operating experience of the CANDU plants from which the CANDU 3 design evolved. AECLT makes no claim of passive shutdown or decay heat removal capabilities. However, because of its unique heavy-water, pressure-tube reactor design and evolution under a different regulatory structure, the CANDU 3 plant does not conform to some current NRC regulations. The staff proposes to apply preapplication review criteria to the CANDU 3 reactor that are consistent with ELWR review requirements.

The staff intends to use the Commission's guidance on these recommendations to conduct preapplication reviews of the conceptual designs. Guidance for review of prototype requirements for advanced reactors will follow SECY-91-074, "Prototype Decisions for Advanced Reactor Designs," March 19, 1991. Consistent with the requirements of Title 10 of the Code of Federal Regulations (CFR) Section 52.47(b)(2), the staff will require that novel safety features of the advanced reactors and CANDU 3 be demonstrated through analysis, test programs, experience, or a combination of these methods. Feedback from the review process will be factored into recommended revisions to the policy guidance, and recommendations for the development of licensing criteria and regulations will be made after the preapplication safety evaluation reports (PSERs) are issued. Should additional issues be developed during the preapplication review process, they will be identified in Commission papers.

The staff is reassessing the priorities and schedules of preapplication reviews in light of staff reductions requested by the Clinton administration. A separate Commission paper is being written by the staff to present the options available for the preapplication reviews of these designs. If the Commission's decision is to stop these reviews, the staff still requests the Commission to review this policy paper, as the Commission's actions on these policy issues will give reactor designers important information for design development.

<u>CONCLUSION</u>:

The staff requests approval of, or alternate guidance on, these proposed positions to be taken in the preapplication review of the advanced reactor and CANDU 3 designs.

<u>COORDINATION</u>:

The Office of the General Counsel has reviewed this paper and has no legal objection to its contents.

<u>RECOMMENDATIONS</u>:

That the Commission

- Approve the staff recommendations in Enclosure 1 for conduct of the preapplication reviews to provide the preapplicants important information for their design development.

- Approve the staff's conclusion that, based on the position that the CANDU 3 design is an evolutionary heavy-water design deriving from CANDU designs operating in Canada and elsewhere, a prototype CANDU 3 is not required for design certification.

- Note that positions which change as preapplication review experience is obtained will be communicated to the Commission and that as the staff identifies new issues it will inform the Commission.

- Note that because of the preliminary nature of the design information on the advanced reactor and CANDU 3 designs, and the preliminary nature of the staff's preapplication reviews, the staff does not recommend proceeding with generic rulemaking on any of the policy issues raised in this paper. The staff will consider generic rulemaking, as appropriate, as the reviews progress and as the staff gains greater confidence in the final design information.

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Enclosures:
1. Analysis of Policy Issues
2. Design Summaries
3. List - Reference Documents
4. Comments - Preapplicant/Other Party w/5 attch.
5. Response to ACRS

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Friday, April 23, 1993.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Friday, April 16, 1993, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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ANALYSIS OF POLICY ISSUES

As part of its preliminary review of the PRISM, MHTGR, CANDU 3, and PIUS designs, the staff has noted 10 instances of the preapplicants proposing to deviate from current light-water reactor (LWR) guidance for the review of the designs. Deviation was proposed when the existing regulations were not applicable to the technology or when the preapplicant considered deviation warranted on the basis of the reactor design and proposed criteria. The staff has grouped the issues into two categories: (1) those issues for which the staff agrees that departures from current regulations should be considered and (2) those issues for which the staff does not believe a departure from current regulations is warranted at this time. The following is a matrix of the issues identifying the plant applicability:

CATEGORY	ISSUES	PRISM	MHTGR	CANDU 3	PIUS
1	A. Accident Evaluation	X	X	X	X
	B. Source Term	X	X	X	X
	C. Containment Performance	X	X	X	X
	D. Emergency Planning	X	X		X
	E. Reactivity Control System				X
	F. Operator Staffing & Function	X	X		X
	G. Residual Heat Removal	X	X		X
	H. Positive Void Reactivity Coefficient	X		X	
2	I. Control Room and Remote Shutdown Area Design	X	X	X	X
	J. Safety Classification of Structures, Systems, & Components		X		

Discussions of these issues follow. The staff has given a brief summary of the issue, current LWR regulations, preapplicant positions, and staff considerations, and proposes a recommendation for staff action. The staff considered the preapplicants' proposals in light of applicable Commission policy statements.

At this preliminary review stage, the staff has limited the scope of the issues to those that could affect the licensability of the proposed design. Additionally, if a similar issue had already been raised for the LWR designs and the staff's advanced reactor design recommendation was essentially the same, it was not repeated in this paper. In those cases where the preapplicants proposed different considerations from the evolutionary or passive LWRs, the issue is treated in this paper in light of the work done in the advanced light-water reactor policy papers.

A. ACCIDENT EVALUATION

Issue

Identify appropriate event categories, associated frequency ranges, and evaluation criteria for events that will be used to assess the safety of the advanced reactor and CANDU 3 designs.

Current LWR Regulations

General Design Criterion (GDC) 4 requires the consideration of accidents in the design basis. Also, 10 CFR 52.47 requires the consideration of consequences for both severe accidents (through the required probabilistic risk assessment) and design-basis accidents (DBAs) for designs that differ significantly from evolutionary designs or utilize passive or other innovative means to accomplish safety functions.

Preapplicant Position

All three advanced reactor preapplicants proposed to analyze accidents that are significantly less probable than the present design-basis range and to assure through their design that these accidents will have acceptable consequences limited to specific dose levels to the public. All chose to utilize the Environmental Protection Agency's (EPA's) lower-level Protective Action Guidelines (PAGs) of 1 rem whole body and 5 rem thyroid as their limits for a significant portion of their accident spectrum. The MHTGR accident guidelines invoke the lower-level PAG dose limit for all sequences more probable than 5×10^{-7} per reactor-year. The PIUS guidelines invoke the PAGs for accident sequences more probable than 10^{-6} per reactor-year. The PRISM guidelines invoke the PAGs for accident sequences more probable than 10^{-4} per reactor-year. The PRISM accident evaluation guidelines also limit consequences from any sequence more probable than 10^{-6} per reactor-year to the 10 CFR Part 100 dose limits. Guidelines for onsite consequences and offsite consequences from operational transients for all vendors are consistent with, or more conservative than, present LWR regulations as contained in 10 CFR Part 100.

The CANDU 3 preapplicant has not submitted detailed analyses of the consequences of design-basis events and severe accidents for the design. The documentation given to the NRC staff indicates that the preapplicant has "considered" design-basis events with estimated frequencies down to 10^{-7} per year and severe accident end-states with estimated frequencies down to 10^{-11} per year. However, the potential consequences to the core, the containment, and the public for any of these events have not been submitted to the staff, except for some limited information for a large loss-of-coolant accident (LOCA) without emergency core cooling, a design-basis accident used to determine the containment design pressure. The preapplicant needs to develop detailed analyses of consequences for anticipated transient without scram (ATWS), unscrammed LOCA, and other events that could cause such reactivity insertions as delayed scram events and control system failures. Because of the positive void reactivity coefficient associated with the CANDU 3 design, events involving even a short scram delay could potentially result in a core disruption accident.

Staff Considerations

The structure proposed by the PRISM, MHTGR, and PIUS preapplicants for selecting accidents to be evaluated was developed to support their positions for reducing emergency planning requirements (as described in Section D of this paper). As discussed in Section D, the staff is not ready to make a specific recommendation on whether, or to what extent, the Commission should consider reducing the emergency planning requirements. The CANDU 3 approach, which limits the scope of severe accidents examined, appears to be inconsistent with the requirements of 10 CFR 52.47; the preapplicant has not documented consequence analyses of design-basis events. Also, no consequences have been analyzed for CANDU 3 severe accidents that would involve substantial core damage. The accident evaluation scheme envisioned by the staff would examine challenging events to the designs to provide information for a later decision on emergency planning requirements and would consider the potential consequences of severe accidents. Additionally, for the multi-module designs (PRISM and MHTGR), the impact of specific events on other reactor modules for the multi-module sites must be assessed.

The staff intends to structure its review conservatively, so that positive conclusions made on the licensability of the conceptual designs during the preapplication review will serve as a reasonable basis for finding the design acceptable at design certification. Some sources of uncertainty regarding the conceptual designs are limited performance and reliability data for passive safety features, lack of final design information, unverified analytical tools used to predict plant response, limited supporting technology and research, limited construction and operating experience, and incomplete quality control information on new fuel manufacturing processes. Later, during the design certification process, some of the conservatism within staff analyses could be removed should completed research lead to improved understanding of the design and better analytical tools.

Staff Recommendation

The staff proposes to develop a single approach for accident evaluation to be applied to all advanced reactor designs and the CANDU 3 design during the preapplication review. The approach will have the following characteristics:

- Events and sequences will be selected deterministically and will be supplemented with insights from probabilistic risk assessments of the specific designs.

- Categories of events will be established according to expected frequency of occurrence. One category of events that will be examined is accident sequences of a lower likelihood than traditional LWR design-basis accidents. These accident sequences would be analyzed without applying the conservatisms used for design-basis accidents. Events within a category equivalent to the current design-basis accident category will require conservative analyses, as is presently done for LWRs.

- Consequence acceptance limits for core damage and onsite/offsite releases will be established for each category to be consistent with Commission policy guidance.

- Methodologies and evaluation assumptions will be developed for analyzing each category of events consistent with existing LWR practices.

- Source terms will be determined as approved by the Commission in Section B of this paper.

- A set of events will be selected deterministically to assess the safety margins of the proposed designs, to determine scenarios to mechanistically determine a source term, and to identify a containment challenge scenario.

- External events will be chosen deterministically on a basis consistent with that used for LWRs.

- Evaluations of multi-module reactor designs will be considered as to whether specific events apply to some or all reactors on site for the given scenario for all operations permitted by proposed operating practices.

B. SOURCE TERM

Issue

Should mechanistic source terms be developed in order to evaluate the proposed designs?

A mechanistic source term is the result of an analysis of fission product release based on the amount of cladding damage, fuel damage, and core damage resulting from the specific accident sequences being evaluated. It is developed using best-estimate phenomenological models of the transport of the fission products from the fuel through the reactor coolant system, through all holdup volumes and barriers, taking into account mitigation features, and finally, into the environs.

Current LWR Regulations

Appendix I to 10 CFR Part 50 (ALARA), 10 CFR Part 100 (Reactor Site Criteria, which references the Technical Information Document (TID) 14844 source term), and 10 CFR Part 20 (Standards for Protection Against Radiation) all have limitations on releases related to power plant source terms.

General Design Criterion (GDC) 60 requires that the design includes means to control suitably the release of radioactive materials in liquid and gaseous effluents and to handle waste produced during normal operations and anticipated operational occurrences.

Preapplicant Position

PRISM designers have proposed calculating a source term different from how the source term is calculated for LWRs. They have proposed siting source terms to bound the release from accidents considered in the design; the magnitude of these source terms is less than the TID-14844 LWR assumed source term. Additionally, at this time, there is insufficient experimental data on the PRISM fuel to quantify the fission product release fractions or the behavior of those fission products migrating from the metal fuel through the sodium coolant.

MHTGR designers have proposed siting source terms for accidents based on the expected fuel integrity. The preapplicant predicts that the coated microsphere fuel particles in the core will contain all the fission products except for products released from the small number of failed particles resulting from in-service particle failures and added particle failures during accidents. Insufficient data currently exist to determine if the MHTGR fuel performance will meet these expectations.

The PIUS designer has proposed using a mechanistic LWR source term. Information has been given in the Preliminary Safety Information Document (PSID) for fission product concentrations in both liquid and gaseous effluents.

The CANDU 3 designer calculates a source term for each scenario. Each accident is evaluated and fission product release and transport are determined individually for each scenario. The staff has not, at this time, evaluated the CANDU 3 codes and methods.

Staff Considerations

In order to evaluate the safety characteristics of advanced reactor designs that are significantly different from LWRs, a different method needs to be developed for calculating postulated radionuclide releases (source terms). In a June 26, 1990, staff requirements memorandum (SRM) related to SECY-90-016, the Commission asked the staff to submit a paper describing the status of efforts to develop an updated source term that takes into account "best available estimates" and current knowledge on the subject. In response to this request, the staff is developing, for LWRs, a revision to the TID-14844 source term (NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," draft report for comment, June 1992).

The differences between the LWR designs and the MHTGR and PRISM designs warrant a separate evaluation of source terms. The CANDU 3 design will also differ from LWR designs in certain respects. The CANDU 3 coolant contains significant amounts of tritium. Following failure of a pressure tube there is no heavy-walled reactor vessel to contain releases (there are large volumes of water in two concentric low-pressure tanks: moderator and shield water). Consequently, the timing of releases is expected to be different from LWRs. However, the same method being used to develop modified LWR source terms should apply to the CANDU 3 design.

The NRC staff is currently revising 10 CFR Part 50 and 10 CFR Part 100 to separate siting from source term dose calculations. The revisions to Part 100 being considered by the staff will replace the present individual dose criteria with a population density standard. A fixed minimum exclusion area radius of 0.4 mile is specified. Other criteria regarding population protection and seismic criteria factors are also included in the source term Part 100 revision. The staff intends that its recommendations for the preapplication review will be compatible with the proposed revisions.

The staff's recommendations envision developing a set of scenario-specific source terms for each of the advanced reactors and CANDU 3 to allow a judgment as to whether the release from each specific sequence meets the accident evaluation criteria for sequences of that event category. Also, a source term may be developed mechanistically for core damage sequences to compare against applicable safety criteria.

Staff Recommendation

Advanced reactor and CANDU 3 source terms should be based upon a mechanistic analysis and will be based on the staff's assurance that the provisions of the following three items are met:

- The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through the research, development, and testing programs to provide adequate confidence in the mechanistic approach.

- The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including specific consideration of containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.

- The events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties.

The design-specific source terms for each accident category would constitute one component for evaluating the acceptability of the design.

C. CONTAINMENT

Issue

Should the proposed advanced and CANDU 3 reactor designs be allowed to employ alternative approaches to traditional "essentially leak-tight" containment structures to provide for the control of fission product release to the environment?

Current LWR Regulations

General Design Criterion (GDC) 16 requires that LWR reactor containments provide an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment, and that containment-associated systems ensure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require. GDC 38, 39, and 40 state requirements for containment heat removal; GDC 41, 42, and 43 for containment atmosphere cleanup; and GDC 50 through 57 for containment design, testing, inspection, and integrity. Requirements for LWR containment leakage testing are established in 10 CFR Part 50, Appendix J.

Preapplicant Position

The MHTGR is not designed with a leak-tight containment barrier. The design relies upon high-integrity fuel particles to minimize radionuclide release, and on a below-grade, safety-related concrete reactor building to provide retention and holdup of any radioactive releases. The reactor vessel and the steam generator vessel are in separate cavities within the concrete structure. In the event of a reactor coolant pressure boundary (RCPB) rupture, louvers in the reactor building are designed to allow the passage of gases to the environment, preventing building overpressure. The building design does not include containment isolation valves for the ventilation line from the building and has an open path to the environment via a drain line in the reactor cavity cooling system (RCCS) panels. Accident dose calculations assume a constant 100-percent volume per day building leak rate, and take credit for plateout on the building walls.

PIUS, above grade, is designed with a low-leakage containment based on a pressure-suppression scheme that is integral with the reactor building, similar to the ABWR and SBWR. Below grade, the bottom part of the containment structure and the monolithic prestressed concrete reactor vessel (PCRV) (which contains the reactor pool water and forms the lower portion of the reactor pressure boundary) are joined together by means of vertical prestressing tendons that are run up to the top of the PCRV. The steel liner on the containment inner surface, which ensures leak-tightness, is a continuous liner covering the whole bottom area, the cylindrical walls, and the upper parts.

CANDU 3 is designed with a large, dry, steel-lined, concrete containment, without containment spray. The maximum leak rate (used in safety analyses) is 5-percent volume per day at the design pressure of approximately 30 psig. The structure is designed for a test-acceptance leak rate of 2 percent per day at the design pressure. These leak rates are significantly higher than leak rates for a typical U.S. pressurized-water reactor (PWR) containment.

The PRISM containment design is a high-strength steel, low-leakage, pressure-retaining boundary, comprising two components, the upper containment dome and lower containment vessel. The upper steel containment dome differs from light-water reactor containments. The containment is specifically designed to mitigate the radioactive release consequences of severe events. The PRISM containment volume is markedly smaller than is typical of LWR containments; there is little separation between the reactor vessel and the containment boundary, and no safety-grade containment coolers or spray systems are provided. The entire containment structure is located below grade within the reactor building.

Staff Considerations

Each of the advanced reactor designs and the CANDU 3 design maintains an accident mitigation approach, part of which includes containment of fission products. Two of the advanced reactor designs (PRISM and MHTGR) place the reactor building below grade, offering protection from external hazards. Generally, the advanced designs focus more attention than do LWRs on protecting the plant by providing passive means of reactor shutdown and decay heat removal (DHR). As a result, designers proposed less-stringent containment requirements.

The staff recognizes that reactor designs without traditional containment structures or systems represent a significant departure from past practice on LWRs, and that existing LWR containment structures have proved an effective component of our defense-in-depth approach to regulation. However, the Advanced Reactor Policy Statement recognizes that to encourage incorporation of enhanced safety margins (such as in fuel design) in advanced reactor designs, the Commission would look favorably on desirable design-related features or reduced administrative requirements. New reactor designs that deviate from current practice need to be extensively reviewed to ensure that a level of safety at least equivalent to that of current-generation LWRs is provided, and that uncertainties in the design and performance are taken into account.

The staff believes that new reactor designs with limited operational experience require a containment system that provides a substantial level of accident mitigation for defense in depth against unforeseen events, including core damage accidents. This requirement may not necessarily result in a high-pressure, low-leakage structure that meets all of the current LWR requirements for containment, but it should be an independent barrier to fission product release. The proposed criteria will need to provide an appropriate level of protection of the public and the environment considering both the safety advantages of the advanced designs and the lack of an experience base in evaluating their performance. For evolutionary LWRs, the staff, in SECY-90-016, proposed to use a conditional containment failure probability (CCFP) or deterministic containment performance goal to ensure a balance and consequence mitigation. During the evolutionary LWR reviews, a great deal of careful review was necessary to assure that a probabilistic CCFP would not be used in a way that could detract from a balanced approach of severe accident prevention and consequence mitigation. For advanced reactor designs and the CANDU 3 design, limited experience exists in the analysis and evaluation of severe accidents which would lead to significant difficulty and uncertainty in assessing a CCFP. For this reason, the staff recommends that the deterministic containment performance goal be adopted for the advanced designs and the CANDU 3 design. The staff proposes to require the preapplicant to postulate a core damage accident as a containment challenge event to demonstrate that containment integrity is maintained for a period of approximately 24 hours after the onset of core damage. This approach is used because the preliminary nature of the advanced designs precludes a reliable assessment of the failure probability of accident mitigation systems and, therefore, of containment failure probability. Further, the CCFP is based on experience with LWR safety systems and accident progression. Intrinsic differences exist between LWR and advanced reactor technologies and their approaches to the balance between accident prevention and mitigation. A quantitative level of understanding of new technologies and systems comparable to that of LWRs is not yet available. Thus, the use of a performance-based criterion rather than a quantitative one appears to be more appropriate for the advanced reactor and CANDU 3 preapplication review given the current level of knowledge of advanced reactor and CANDU 3 risk and their prevention and mitigation elements.

Staff Recommendation

The staff proposes to utilize a standard based upon containment functional performance to evaluate the acceptability of proposed designs rather than to rely exclusively on prescriptive containment design criteria. The staff intends to approach this by comparing containment performance with the accident evaluation criteria.

- Containment designs must be adequate to meet the onsite and offsite radionuclide release limits for the event categories to be developed as described in Section A to this paper within their design envelope.

- For a period of approximately 24 hours following the onset of core damage, the specified containment challenge event results in no greater than the limiting containment leak rate used in evaluation of the event categories, and structural stresses are maintained within acceptable limits (i.e., ASME Level C requirements or equivalent). After this period, the containment must prevent uncontrolled releases of radioactivity.

D. EMERGENCY PLANNING (EP)

Issue

Should advanced reactors with passive design safety features be able to reduce emergency planning zones and requirements?

Current LWR Regulations

Although emergency plans are not required for the issuance of a design certification under 10 CFR Part 52, they would be necessary for the issuance of a combined license under Part 52 or a license issued under 10 CFR Part 50. 10 CFR 50.47 requires that no operating license be issued unless the NRC finds that there is reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

Currently, offsite protective actions are recommended when an accident occurs that could lead to offsite doses in excess of Environmental Protection Agency (EPA) Protective Action Guidelines (PAGs), which are 1-5 rem whole body and 5-25 rem thyroid. At the lower projected doses, protective actions should be considered. At the higher projected doses, protective actions are warranted.

Preapplicant Position

The proposed approach of PRISM designers to emergency planning (EP) is significantly different from that of previous LWR applicants, particularly in the area of offsite EP. An objective of PRISM designers is to meet the lower-level PAG criteria so that formal offsite EP involving early notification, detailed evacuation planning, and provisions for exercise of the plan would not be required. In order to attain this objective, the PRISM design emphasizes accident prevention, long response times (36 hours) between the initiation of an accident and core damage, and the mitigation of releases if they should occur.

The MHTGR designers proposed reduced offsite emergency planning for reasons similar to those proposed for the PRISM design. An emergency plan would be written for an MHTGR and the plan would include any agency that could become involved in the response to a radiological emergency (i.e., sheltering and evacuating the public and controlling the food supply). The differences and reductions from a typical emergency plan for LWRs are that the MHTGR plan would have the exclusion area boundary (EAB) of 10 CFR Part 100 as the boundary of the emergency planning zone (EPZ), as may be allowed by Appendix E of 10 CFR Part 50 for gas-cooled reactors; and that there would be no rapid notification or annual drills for offsite agencies. This is based on the preapplicant's assertion that (1) the predicted dose consequences estimated at the EAB/EPZ for accidents are below the lower-level EPA sheltering PAGs and that the public can be excluded from the EAB, (2) there is a significantly long time expected for the core to return to criticality after being shut down by the Doppler coefficient without the reactor protection system functioning (i.e., about 37 hours), and (3) there is a long time for the fuel and reactor vessel to reach maximum temperatures (i.e., about 100 hours) during accidents.

The preapplicant asserts that the public around the plant would be outside the area that needs to be sheltered or evacuated and, further, there will be ample time to notify and move the public during an event.

With regard to the PIUS design, the preapplicant expects that due to its passive safety features, onsite and offsite emergency planning will be considerably simplified in comparison with current day LWRs. The preapplicant believes that regulatory relief could be considered based on (1) satisfaction of the EPA lower-level PAGs for dose, (2) the passive features of the PIUS design which precludes core damage, and (3) design features that allow substantially larger response times before intervention would be required. The preapplicant contends that there are no credible accident sequences that would lead to severe core damage. Offsite dose for the large-break LOCA is claimed to be below the lower-level EPA PAGs at 500 meters distance from the containment.

Staff Considerations

The designers of advanced reactors have objectives of achieving very low probabilities ($<1.0x10^{-6}$ per year) of exceeding the EPA lower-level PAGs. The designers claim that these advanced reactors, with their passive reactor shutdown and cooling systems, and with core heatup times much longer than those of existing LWRs, are sufficiently safe that the EPZ radius can be reduced to the site boundary, and that detailed planning and exercising of offsite response capabilities need not be required by NRC regulation. The preapplicants state that this does not mean that there would be no offsite emergency plan developed, but rather that such a plan could have fewer details concerning movement of people, and need not contain provisions for early notification of the general public or periodic exercises of the offsite plan on a scale consistent with present reactors.

A similar policy issue was identified for the passive LWR design, but remains open. The Electric Power Research Institute (EPRI) is working with the NRC staff to define a process for addressing simplification of emergency planning. The results of this effort should be applicable to advanced reactor designs.

Staff Recommendation

The staff proposes no changes to the existing regulations governing EP for advanced reactor licensees at this time. The staff will provide regulatory direction at or before the start of the design certification phase so that any EP implications on design can be addressed. Consistent with the current regulatory approach, the staff views the inclusion of emergency preparedness by advanced reactor licensees as an essential element in NRC's "defense in depth" philosophy. Briefly stated, this philosophy (1) requires high quality in the design, construction, and operation of nuclear plants to reduce the likelihood of malfunctions in the first instance; (2) recognizes that equipment can fail and operators can make mistakes, thus requiring safety systems to reduce the chances that malfunctions will lead to accidents that release fission products from the fuel; and (3) recognizes that, in spite of these precautions, serious fuel damage accidents can happen, thus requiring containment structures and other safety features to prevent the release of fission products off site. The added feature of emergency planning to the defense-in-depth philosophy provides that, even in the unlikely event of an offsite fission product release, there is reasonable assurance that emergency protective actions can be taken to protect the population around nuclear power plants.

Information obtained from accident evaluations conducted as outlined in Section A of this paper will be factored into to the emergency planning requirements for advanced reactor designs. Based in part upon these accident evaluations, the staff will consider whether some relaxation from current requirements may be appropriate for advanced reactor offsite emergency plans. The relaxations to be evaluated will include, but will not be limited to, notification requirements, size of EPZ, and frequency of exercises. This evaluation will take into account the results of passive LWR emergency planning policy decisions.

E. REACTIVITY CONTROL SYSTEM

Issues

Should the NRC accept a reactivity control system design that has no control rods?

Current LWR Regulations

General Design Criterion (GDC) 26 requires that a nuclear plant have two independent reactivity control systems. One of the systems shall use control rods, and shall preferably use a positive means for inserting them. The other system shall be capable of controlling planned reactivity changes to ensure that fuel limits are not exceeded.

Preapplicant Position

The PIUS design has no control rods. However, the preapplicant proposes that the design complies with the intent of GDC 26 by having two independent liquid boron reactivity control systems. The normal reactivity control system pumps boron into the primary coolant loop to control reactor power or effect a reactor shutdown; this system is only safety grade within the bounds of the containment isolation valves. This system recently has been modified by the inclusion of lines between the reactor pool and the inlet to the reactor coolant pump in each loop. These lines contain two "scram" valves, in parallel, that are normally closed. Upon receipt of a scram signal, the valves open and the imposed pressure difference induces highly borated water from the pool to flow to the pump inlet. The pumps continue to operate, thereby injecting highly borated water into the primary circulation loop. The fully safety-grade reactivity shutdown control system relies on the ingress of highly borated water through the density lock from the reactor pressure vessel to scram the reactor. This ingress occurs when the hydraulic equilibrium conditions across the density locks are disturbed. Either a trip of as few as one of the four reactor coolant pumps or a reactor overpower event (with forced flow) could initiate borated water flow through the lower density lock and into the core. The scram valve system modification provides a redundant rapid shutdown mechanism which is not precluded by blockage of the density lock(s). Other reactivity control features of the design are in-core burnable poisons for power shaping, and limitations in core size for control of xenon oscillations for slow, large, and small reactivity changes. For rapid changes, the design relies on the highly negative moderator temperature coefficient of reactivity.

The density locks, essentially bundles of open, parallel tubes about 2.25 inches in diameter, have no moving parts. They are of safety-grade construction and intended to be highly reliable. However, their function must be demonstrated, and the potential for and the effects of blockage and high-cycle thermal fatigue cracking must be evaluated.

Staff Considerations

The existing LWR regulations provide prescriptive design guidance for one reactivity control system to use control rods. Of the three advanced reactor designs, only PIUS does not have the capability to control reactivity with control rods. The PIUS design does have, however, three ways to introduce liquid boron into the core to control and shut down the reactor. Two of the three rely on a common supply of borated pool water: one through the density locks and the other through the scram valves. The other system is the normal reactivity control system which has a separate boron tank and is used for normal shutdown. The latter system is only safety grade within the bounds of the containment isolation valves.

Staff Recommendation

The staff concludes that a reactivity control system without control rods should not necessarily disqualify a reactor design. A design without control rods may be acceptable, but the preapplicant must provide sufficient information to justify that there is an equivalent level of safety in reactor control and protection as compared to a traditional system that has rods. This information must include the areas of

- reliability and efficacy of scram function
- suppression of oscillations
- control of power distribution
- shutdown margin
- operational control

F. OPERATOR STAFFING AND FUNCTION

Issue

Should advanced reactor designs be allowed to operate with a staffing complement that is less than that currently required by the LWR regulations.

Current LWR Regulations

The NRC established the requirements for control room staffing in 10 CFR 50.54(m)(2)(iii). That regulation states that a senior operator must be present in the control room at all times and a licensed operator or senior operator must be present at the controls of a fueled nuclear power unit. In 10 CFR 50.34(m)(2)(i) a table lists the minimum staffing requirements for an operating reactor.

Standard Review Plan Section 13.1.2, Paragraph II.C states that at any time a licensed nuclear unit is being operated in modes other than cold shutdown, the minimum shift crew shall include two licensed senior reactor operators (SROs), one of whom shall be designated as the shift supervisor, two licensed reactor operators (ROs), and two unlicensed auxiliary operators (AOs).

Preapplicant Position

The MHTGR plant has four reactor modules, with two modules feeding a single steam supply system. The design includes a shift staffing level of eight persons who are dedicated to plant operations: a senior licensed shift supervisor, two licensed reactor operators in the control room, and five roving non-licensed operators. This adds up to three licensed and five non-licensed operators for four reactor modules.

The PRISM control room would contain the instrumentation and controls for up to nine reactor modules and their power conversion systems. The minimum number of operating staff would include one SRO shift supervisor, one SRO assistant supervisor, one RO per power block (three modules) in the control room, and three plant ROs. This adds up to a minimum of eight licensed operators for nine reactor modules.

During normal plant operations, the PIUS main control room would be manned by two ROs and an SRO shift supervisor. The shift supervisor would not be required to be in the control room at all times.

Staff Considerations

Present-day LWRs are required to have a minimum of one shift supervisor, one SRO, and two operators per reactor. The designers of advanced reactors have stated that the highly automated operating systems, the passive design of safety features, and the large heat capacity result in reactor designs that respond to transients in a manner that demands less of the operator than do currently operating plants or evolutionary designs. The advanced reactor preapplicants assert that because of passive safety features and, in some cases, large coolant inventories, operator actions may not be required for several days following an accident. These designs also automate systems that start up, shut down, and control these reactors. The designers of advanced reactors suggest that they could be operated with fewer licensed operators and that this would reduce the training and operating costs to licensees significantly. The CANDU 3 preapplicant has not proposed a specific number of licensed operators, but the staff expects that CANDU 3 will meet the current LWR staffing requirements.

A similar policy issue, the role of the operator in a passive plant control room, was discussed in the staff's June 25, 1992, draft policy paper on passive reactors. In that paper, the staff expressed concern that the man-machine interface for the passive reactors had not been sufficiently addressed and that actual testing needed to be done on a control room prototype. The staff believes that position is also applicable to advanced reactors.

Staff Recommendation

The staff believes that operator staffing may be design dependent and intends to review the justification for a smaller crew size for the advanced reactor designs by evaluating the function and task analyses for normal operation and accident management. The function and task analyses must demonstrate and confirm the following through test and evaluation:

- Smaller operating crews can respond effectively to a worst-case array of power maneuvers, refueling and maintenance activities, and accident conditions.

- An accident at a single unit can be mitigated with the proposed number of licensed operators, less one, while all other units could be taken to a cold-shutdown condition from a variety of potential operating conditions, including a fire in one unit.

- The units can be safely shut down with eventual progression to a safe shutdown condition under each of the following conditions: (1) a complete loss of computer control capability, (2) a complete station blackout, or (3) a design-basis seismic event.

- The adequacy of these analyses shall be tested and demonstrated. The staff is currently recommending that an "actual control room prototype" be used for test and demonstration purposes.

G. RESIDUAL HEAT REMOVAL

Issue

Should advanced reactor designs that rely on a single completely passive, safety-related residual heat removal (RHR) system be acceptable?

Current LWR Regulations

General Design Criterion (GDC) 34 requires the RHR to function using only safety-grade systems, assuming a loss of either onsite or offsite power, and assuming a single failure within the safety system. Regulatory Guide 1.139 (issued in draft for comment), augmenting GDC 34, states that the RHR function should be capable of bringing the plant to a safe shutdown condition within 36 hours of reactor shutdown. Branch Technical Position (BTP) RSB 5-1 states that the RHR function must be performed in a reasonable period of time following reactor shutdown.

Preapplicant Position

The PRISM design uses the reactor vessel auxiliary cooling system (RVACS) as the safety-grade system for removing residual heat from the reactor core. Reactor-generated heat is transferred through the reactor vessel to the outer surface of the containment vessel. RHR is then accomplished by means of heat transferring to the atmosphere through natural circulation. Cooler air flows downward into the below-grade reactor silo, where it is turned inward and upward to be heated by the outer surface of the containment vessel and a special collector cylinder. This heated air then flows out of the silo and is released to the atmosphere. The RVACS is completely passive and always in operation. The RVACS is proposed as a backup to normal non-safety-grade cooling through the intermediate heat transport system, the steam generator, and condenser. If the condenser is not available for cooling but the intermediate sodium loop remains available, then the non-safety-grade auxiliary cooling system (ACS) supplements the RVACS. The ACS operates through natural circulation air cooling of the steam generator. The RVACS design-basis analysis (performed by the designer) results in high-temperature conditions (but within design limits) for an extended period of time if no other system is operated. However, use of the ACS in conjunction with the RVACS can limit peak coolant temperature for decay heat removal to about 15 °C above normal operating temperatures.

The MHTGR is designed with only one safety-grade system for removing residual heat from the core, the reactor cavity cooling system (RCCS). The RCCS consists of panels within the reactor cavity and ducts connecting the RCCS panels to four inlet/outlet ports. Redundancy is provided by these separate ports and a cross-connected header that surrounds the reactor vessel (i.e., any panel can be fed from any inlet and can discharge to any outlet). The RCCS operates by absorbing radiant heat from the reactor vessel to the panels that surround the reactor vessel and transferring the heat by convection to the air flowing by natural circulation in the panels. As the heated air rises, cooler atmospheric air is drawn to the panels through the inlet ports.

There are no active components in the RCCS. The system is always in operation and cannot be turned off. Instrumentation can monitor the performance of the RCCS. The RCCS is relied upon should the non-safety-grade heat transport system (HTS) and the shutdown cooling subsystem (SCS) be inoperable. The HTS utilizes the steam generators and non-safety-grade feed system and condensers and is used during normal operations, startup/shutdown, and refueling. The SCS is a backup to the HTS and uses an alternate helium circulator for core cooling and an additional heat sink, the shutdown cooling heat exchanger. The use of the non-safety-grade backup RHR systems reduces the frequency, magnitude, and duration of high-temperature challenges to the reactor vessel, and the slow time scale (days) of MHTGR core heatup events allows time to bring the non-safety-grade systems back into service.

The PIUS design uses a safety-grade passive closed cooling system (PCCS) for removing residual heat from the reactor pool. The system consists of eight independent parallel loops located in four separate compartments that are physically separated from each other. Heat is dissipated through four natural draft cooling towers physically separated from each other and located on the top of the reactor building. One cooling tower is in each quadrant of the reactor building. The reactor pool water can be maintained at 95 °C with one loop out of service. The system is always in operation. Residual heat can be removed from the reactor by means of the condenser during startup/shutdown and refueling conditions. If the condenser is not available, a non-safety-grade diesel-backed pump system can cool the pool water.

Staff Considerations

Similar issues were identified for the RHR system of the passive LWR designs. In a draft Commission paper issued for comment on February 27, 1992, the staff discussed issues relating to the (1) ability of passive systems to reach safe shutdown, (2) definition of a passive failure, and (3) treatment of non-safety systems that reduce challenges to the passive systems. These issues have not been resolved and the staff will propose recommendations for resolving them.

In the case of advanced reactors, the safety-grade RHR systems are completely passive and are in continuous operation. Continuous performance monitoring of the passive systems is one advantage of the constant operation. The high heat capacity of the PRISM and MHTGR designs lead to longer time periods before exceeding temperature limits. PRISM and MHTGR use the natural circulation of air to remove residual heat. PIUS uses natural circulation of water through natural draft cooling towers for its RHR system. The lack of check and squib valves, and the continuous operation and use of a single-phase fluid in the PIUS system appear to offer increased reliability over the passive LWR systems.

However, relying solely on passive systems may lead to high-temperature challenges to the reactor vessel and reactor internal structures, because higher heat removal rates are needed for passive cooling situations which require larger temperature differences between the reactor and cooling medium (air). Elevated temperatures (above normal operating values) may exist in the vessel and internal structures for long periods of time. In the high-temperature reactors, the PRISM and MHTGR, creep damage may be more likely as the result of these long-term high-temperature transients.

Staff Recommendation

The unique features of the PRISM, MHTGR, and PIUS designs, lead the staff to believe that reliance on a single, completely passive, safety-related RHR system may be acceptable. In performing its detailed design evaluation, the staff will ensure that NRC regulatory treatment of non-safety-related backup RHR systems is consistent with Commission decisions on passive LWR design requirements.

H. POSITIVE VOID REACTIVITY COEFFICIENT

Issue

Should a design in which the overall inherent reactivity tends to increase under specific conditions or accidents be acceptable?

Current LWR Regulations

General Design Criterion (GDC) 11 requires that the reactor core and coolant system be designed so that in the power operating range the net effect of prompt inherent nuclear feedback characteristics tend to compensate for rapid increases in reactivity.

Preapplicant Position

In the PRISM design, the maximum sodium void worth, according to the preapplicant, assuming only driver fuel and internal blanket assemblies void, is nominally \$5.50. If radial blanket assemblies are included, the sodium void worth is nominally \$5.26, which does not include the -70¢ from the gas expansion modules (GEMs). Should sodium boiling begin on a core-wide basis, assuming failure to scram conditions with a total loss of flow without coastdown, the reactor could experience a severe power excursion and core disruption. The predicted temperature reactivity feedback is approximately -80¢ preceding the onset of sodium voiding. This mitigates to some extent the positive reactivity addition. For sodium voiding to occur, redundant and diverse safety-grade systems would have to experience multiple failures.

Although the preapplicant claims that the overall power coefficient for a CANDU 3 reactor is slightly negative and very close to zero, the coolant void reactivity is positive throughout the fuel core lifetime. The total core void worth is between \$1 and \$2. The positive void coefficient is not a concern during normal operation, but, during a large LOCA at specific locations, void reactivity increases dramatically. If CANDU 3 were to experience a large-break LOCA (guillotine rupture of an inlet header) with a failure of both safety-grade shutdown systems, the positive void reactivity insertion could lead to a power excursion followed by core melting. The CANDU 3 design is intended to prevent an unscrammed event from occurring through the use of two separate shutdown systems, each to be independent, redundant, diverse, and safety grade.

Because of intrinsic design characteristics, both the MHTGR and the PIUS designs have overall negative void and moderator reactivity coefficients at operating conditions throughout the burnup cycle. Although the PIUS design relies on borated water to achieve shutdown, instead of control rods as in contemporary LWR designs, the displacement of this water during voiding, which introduces positive reactivity, is offset by negative reactivity, which occurs because of the attendant decrease in neutron macroscopic scattering cross section.

Staff Considerations

The existence of positive coolant void coefficients, or any reactivity effect that tends to make a postulated accident more severe, is a significant concern. As a result of a positive void reactivity coefficient, events involving even a relatively short scram delay could lead to a core disruption accident. The staff intends to require the preapplicant to analyze the consequences of events (such as ATWS, unscrammed LOCAs, delayed scram events, and transients that affect reactivity control) that could lead to core damage as a result of the positive void coefficient, taking into account the overall risk perspective of the designs. A core disruption accident in either the PRISM or CANDU 3 designs may not necessarily lead to a large-scale release of the radionuclide inventory to the atmosphere due to their respective mitigative designs. In the CANDU 3 reactor, multiple, redundant, diverse, fast-acting scram systems address the positive coefficients.

Attempts to modify the designs in order to reduce the effects of these positive coefficients may result in other consequences potentially as serious. For example, in the PRISM design, the positive void coefficient seems to result from the design objectives of maintaining a passive shutdown capability and of minimizing the reactivity swing over core life. Attempts to reduce the PRISM void worth might have the effect of increasing the severity of rod withdrawal accidents or reducing the ability to withstand an unscrammed loss of heat sink event without core damage.

Staff Recommendation

The staff concludes that a positive void coefficient should not necessarily disqualify a reactor design. The staff is proposing to require that the PRISM and CANDU 3 preapplicants analyze the consequences of events (such as ATWS, unscrammed LOCAs, delayed scrams, and transients affecting reactivity control) that could lead to core damage as a result of the positive void coefficients. When it reviews these analyses, the staff will take into account the overall risk perspective of the designs. Whether the preapplicants will be required to consider changes in the designs to mitigate the consequences of these accidents will depend on the estimated probability of the accidents as well as the severity of the consequences.

I. CONTROL ROOM AND REMOTE SHUTDOWN AREA DESIGN

Issue

Can current requirements for a seismic Category I/Class 1E control room and alternate shutdown panel be fulfilled by a remote shutdown area, and a non-seismic Category I, non-Class 1E control room?

Current LWR Regulations

The current LWR requirements for control room and remote shutdown area design are given in 10 CFR Part 50, Appendix A, and 10 CFR Part 100. General Design Criterion (GDC) 19 requires that a control room with adequate radiation protection be provided to operate the plant safely under normal and accident conditions and that there be an ability to shut down the plant from outside the control room. GDC 17 requires that the electrical system for the control room and remote shutdown equipment meet the requirements for quality and independence. These requirements are defined as Class 1E in the supporting IEEE standards. GDC 2 and 10 CFR Part 100 require that structures and systems important to safety be designed to seismic Category I standards to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

The current LWR acceptance criteria and guidelines for the remote shutdown area(s) are given in Standard Review Plan (SRP) Section 7.4. The SRP states that the area(s) should be separate from the control room as, for example, local control panels. These area(s) should be in communication with the control room, and should have Class 1E monitoring instrumentation and controls capable of bringing the reactor down to cold shutdown, as well as, be designed to meet single-failure criteria and seismic Category I.

Preapplicant Position

The control room for the PRISM reactor contains the instrumentation and controls for all nine reactor modules and their power conversion systems. The control room structure is not considered safety related and, therefore, the room is not designed to seismic Category I design requirements. Additionally, no equipment in the control room is safety grade. A separate alternate shutdown console is located in the protected area of the reactor service building. The alternate shutdown console is within a seismic Category I structure and is equipped with the necessary Class 1E controls and instrumentation to protect the core and has the required habitability system.

The MHTGR design for the four reactor modules has a non-safety-related central control room to operate the plant and a seismic Category I remote shutdown area from which to respond to accidents if necessary. Neither the equipment in the control room nor the remote shutdown area are Class 1E. The remote shutdown area does not contain safety-related equipment, nor does it include a ventilation system for operator habitability, or a safety-related manual scram. This is based on the preapplicant's position that accidents do not require operator response. The only manual scrams are non-safety-related and are located in the remote shutdown area and the main control room. The plant also has a separate reactor protection system vault in each reactor module which has Class 1E instrumentation and controls for the reactor, and is seismic Category I. The preapplicant has not stated that the vault has a manual scram.

The CANDU 3 design utilizes a main control room to perform all monitoring and control functions for normal operation and most accident conditions. The preapplicant states that the main control room is not designed to provide safety-grade functions following a design-basis earthquake, tornado, fire, or loss of non-essential (i.e., Class IV) Group 1 electrical power, but the operator is able to proceed to the secondary control area. The secondary control area duplicates the control consoles in the main control room for the control and monitoring of Group 2 systems only. Safety-related systems for the CANDU 3 design are found in both Group 1 and Group 2 designated systems, which implies that some Group 1 safety-related functions might not be accomplished from the secondary control area. The secondary control area is seismically qualified and is electrically isolated from the main control room so that failures occurring in the Group 1 area will not interfere with control and monitoring of safety systems from the secondary control area. All equipment and structures making up the route from the main control room to the secondary control area are to be qualified to the extent necessary to prevent route blockage, fire, or flood. CANDU 3 has specified requirements to ensure habitability during accident conditions.

The central control room for the PIUS design is a seismic Category I structure. However, the safety-related systems within this structure are for monitoring only to assure that the core is protected. Although the operator could take actions, these actions would be taken with the use of non-safety-grade controls. The two remote shutdown areas are housed in separate compartments at the bottom of the reactor building in protected seismic Category I areas. Each remote area contains one-half of the safety-grade control equipment (the reactor trip and interlock system, control of certain isolation valves, and safety-grade monitoring systems). The manual reactor trip system is a push-button control of the scram valves. Both the main control room and the emergency shutdown areas are serviced by a safety-grade ventilation system to ensure habitability during accidents.

Staff Considerations

The staff believes that the operators remain a critical element in ensuring reactor plant safety and that no increased burden should be placed on operators managing off-normal operations. The control room is the space in the plant where operators are most familiar with the surroundings and normally manage plant activities. The staff is reluctant to approve any design that would increase the frequency of evacuation of the control room during design-basis accident conditions or hamper the control or monitoring of upset conditions as an event sequence progresses. The staff believes human performance will still play a large role in the safety of the advanced plants and the CANDU 3 plant and that the quality of support provided by a safety-related, seismic Category I and electrical Class 1E control room is appropriate.

The staff also believes that any remote shutdown area should be designed to complement the main control room. Sufficient Class 1E instrumentation and controls should be available to effectively manage anticipated accidents that would cause a loss of the control room functions. The location and qualification of the remote shutdown areas should also ensure protection of the remote shutdown operations to the greatest extent possible.

A related policy issue was identified in the staff's February 27, 1992, draft paper on policy issues for the passive LWRs. EPRI proposed less-conservative control room habitability requirements and proposed that analyses of control room habitability be limited to 72 hours instead of to the accident duration. The staff disagreed with the proposed EPRI guidance and offered different criteria. Similarly, the staff, in its June 25, 1992, draft policy paper, defined positions on common-mode failures in digital systems and on annunciator reliability. Staff requirements for advanced reactor designs will be consistent with passive LWR policy guidance for these issues, once the guidance is finalized.

Staff Recommendation

Until passive LWR policy for design requirements of control rooms and remote shutdown facilities is determined, the staff will apply current LWR regulations and guidance to the review of advanced reactor designs. This will ensure that plant controls and the operators will be adequately protected so that safe shutdown can be assured in accident situations.

J. SAFETY CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

Issue

What criteria should the NRC apply to the advanced reactor designs to identify the safety-related structures, systems, and components?

Current LWR Regulations

Title 10 of the Code of Federal Regulations Section 50.49(b)(1) and the current Appendix A.VI(a)(1) of 10 CFR Part 100 list the following criteria to identify the safety-related structures, systems, and components (SSCs):

(a) those needed to maintain the integrity of the reactor coolant pressure boundary (RCPB)

(b) those needed to shut down the reactor and maintain it in a safe condition

(c) those needed to prevent or mitigate the consequences of accidents that could result in doses comparable to Part 100 guidelines

These criteria require that a reactor design protect the public from the potential release of fission product radioactivity by means of three barriers: the reactor coolant pressure boundary (item a above), the fuel (item b above), and the containment (item c above).

Amendments to Parts 50 and 100 have been proposed (57 FR 47862) to update criteria used in decisions regarding reactor siting and design for future nuclear power plants, including the advanced LWR designs. These proposed revisions include the temporary relocation of the dose considerations for reactor siting (i.e., the current Part 100 guidelines) from Part 100 to Part 50 until such time as more specific requirements are developed regarding accident source terms based on insights into severe accidents.

Preapplicant Position

The advanced reactor designs rely on a limited number of safety-related systems to protect the core and the public. Some of these systems are entirely passive, have no moving components and do not require operator action. The designers believe that this reduction in safety-related equipment results in simpler plant designs with lower costs. This also results in many structures, systems, and components, which are considered as safety related in LWR designs, being classified as non-safety-related in the advanced reactor designs.

Of the advanced reactor designs, only the MHTGR design is not using the current LWR criteria for safety classification. For the MHTGR design, the only criterion for safety-grade classification is those structures, systems, and components needed to mitigate the dose consequences at the site boundary from accidents or events to a dose consequence which is below the guidelines in the current 10 CFR Part 100. The staff described several major issues concerning safety classification in the draft PSER (NUREG-1338): (1) the RCPB is not entirely safety related, (2) no safety-related equipment is used to pressurize and depressurize the RCPB, (3) the coolant moisture monitor is not safety related, (4) neither the control room nor remote shutdown area are safety related, and (5) no safety-related instrumentation providing reactor protection or monitoring functions are available in the control room or remote shutdown area. The safety classification criteria proposed by the MHTGR preapplicant requires, in effect, the protection of only one barrier to the potential release of fission product radioactivity to the public: the fuel.

Staff Considerations

The NRC LWR safety classification criteria are based on the fundamental regulatory standard to require defense in depth for a reactor design and to require safety-related SSCs to separately protect the three barriers to potential releases of fission product radioactivity to the public: the fuel, the reactor coolant pressure boundary, and the containment. This approach by definition requires that safety-related SSCs be identified to protect more than just one of the traditional barriers, e.g., more than just the fuel barrier to radionuclide transport.

The advanced reactor designs include high-quality, non-safety-related active systems to provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. These would be the first line of defense should transients or other plant upsets occur. The non-safety-related systems are, according to the designers, not required for mitigation of design-basis events, but do provide alternate mitigation capability. In a recent draft SECY paper covering the passive ALWRs, the NRC staff stated that it was still evaluating the issue of treatment of non-safety-related systems for the passive ALWRs and that the proposed resolution to this issue would be provided later. The staff plans to treat non-safety-related systems consistent with the eventual position for passive LWRs.

Staff Recommendations

The staff intends to apply current LWR criteria for the identification of safety-related SSCs to the MHTGR design. The staff will consider arguments from the MHTGR preapplicant for reducing the design, installation, and maintenance requirements of the identified safety-related SSCs for the MHTGR design. Requirements for non-safety-related systems will be consistent with the NRC position for passive LWRs. The staff has noted that LWR criteria may be restructured within Parts 50 and 100, and our expectation is that the criteria in Part 50 will apply to the standard design certification.

ATTACHMENT 1

AECLT SPECIFIC COMMENTS ON CANDU 3 DESIGN ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

In this attachment, AECLT addresses the six issues which the draft Policy Issues Paper identified as pertaining to the CANDU 3 design; specifically, Accident Evaluation, Source Term, Containment Performance, Operator Staffing, Positive Void Reactivity and Control Room Design. Additionally, AECLT comments on the effects of the proposed Category 2 classification issues.

For each of the six issues, the draft Policy Issues Paper characterizes and discusses AECLT's approach to the CANDU 3 design. Based on the characterization and discussion, the paper proposes a recommended resolution of the issue. AECLT believes that the Staff's current evaluation does not give credit to the CANDU design approach.

1. The basis for ATWS in the first place was the single line of defense in LWRs against some accidents. CANDU chose to address the ATWS question by having redundant shutdown. The accidents listed by the Staff are not excluded from consideration in CANDU, but have little consequence because of the redundant shutdown systems. The whole point of redundant shutdown is to provide real safety, as opposed to providing analysis of events without shutdown. This recognition is lacking.

2. Events which would lead to core melt in conventional LWRs, namely LOCA/LOECC, do not do so in CANDU because of the presence of the moderator. To use the consequences of a severe accident to challenge the design, without examining the defenses that have to fail before those consequences occur, removes the incentive from the designer to reduce the frequency of those consequences.

AECLT corrects the characterization and the evaluation in the discussion section, as necessary. In addition, AECLT comments on the recommended resolution of the issue and, where AECLT differs with the recommendation, offers an alternative approach for consideration.

A. ACCIDENT EVALUATION

ISSUE: Identify appropriate event categories, associated frequency ranges, and evaluation criteria for events that will be used to assess the safety of the proposed designs.

AECLT COMMENT: AECLT does not believe that the draft Policy Issues Paper accurately characterizes AECLT's approach to accident evaluation. The draft paper states:

> "The CANDU 3 preapplicant, in their current safety analyses, has excluded analyses of the consequences of events with frequencies of less than 10^{-6}/year from the safety evaluation. Events which would be excluded from consideration, based on the CANDU 3 design characteristics and system reliabilities, would include anticipated transient without scram (ATWS), unscrammed loss-of-coolant-accidents (LOCAs), delayed scram events, and other events which could affect reactivity insertion (for example, from control system failures). As a result of the positive void reactivity coefficient associated with the CANDU design, events involving even a relatively short scram delay could result in a core disruption accident."

AECLT's approach, which is based on design review guides accepted by the Atomic Energy Control Board (AECB), is summarized in the points which follow. We request that the draft paper be revised in accordance with these points.

1. For the CANDU 3 design evaluation, event sequences and their End States are determined by systematic review without regard to End State frequencies. The Conceptual Probabilistic Safety Assessment (CPSA) considers End State frequencies as low as 10^{-11}/year.

2. Reactivity insertion events are not excluded from consideration. Three systems are provided for such events:

    1) The Group 1 Regulatory System with Mechanical Control Absorber Rods for Anticipated Transients.

    2) The Group 2 Shutdown System 1 with rapid shutdown rods for Accidents.

    3) The Group 2 Shutdown System 2 with rapid liquid poison injection for Accidents.

    The CPSA gives the end state frequency for the large LOCA with a failure to shutdown at 10^{-10}/year.

3. The CANDU 3 Safety Analysis has no absolute frequency cutoff. As stated in the Conceptual Safety Report, Appendix C, Section 5.2 Category B Events:

    > "The events in Category B are those for which the frequency of the event can be calculated using probabilistic tools to obtain a realistic assessment of the risk involved (public and economic risk)...

> "The stepped curves in Figure 2 will be used as the acceptance criteria for Category B analyses. These are intended to be used as event-based criteria, to provide a measure of the acceptability of the consequences of a given event, which is a function of its likelihood of occurrence. As in the previous probabilistic assessments, events with frequencies less than 10^{-6} events per year are not considered to be of high enough frequency that they generally need to be considered. <u>In those cases where they are, Figure 2 will be extrapolated as necessary.</u>"

4. The current CANDU 3 Safety Analysis focuses primarily on identifying design requirements and recommendations for design improvement and assessing the adequacy of the safety systems. The Safety Analysis establishes categories of events, along with evaluation methods and acceptance criteria for each event.

5. The events analyzed are selected because of their impact on the conceptual design, regardless of their frequency. For example, for the containment design, the event analyzed is a large loss-of-coolant accident with emergency core cooling unavailable (LOECC).

   With respect to the treatment of ATWS, unscrammed LOCAs and delayed scram events, see the discussion below in the section concerning Positive Void Reactivity.

6. The draft Policy Issues Paper also states:

   > "The CANDU 3 approach which limits the scope of severe accidents examined appears to be inconsistent with the provisions of 10 CFR 52.47."

   This statement is not accurate. As discussed above in Point No. 3, limits are not placed on the scope of severe accidents that may be considered in designing the CANDU 3.

7. In the draft Policy Issues Paper, the Staff proposes to develop "a single approach to all advanced reactor designs during the preapplication review." Although omitting mention of its applicability to the CANDU 3 design, it appears that the approach is to apply to the CANDU 3 design as well. Assuming that to be the case, the paper should be corrected.

8. The first bullet in the Recommendations section states:

   > "Events will be selected deterministically and supplemented with insights from probabilistic risk assessments of the specific designs."

   AECLT believes that the criteria which will be used in deterministically selecting the events should be identified. Also, we would like to know whether the Staff will continue the historical requirement for conservative analysis for Design Basis Accidents (DBAs) and best estimate analyses for beyond DBAs. The NRC Staff recommendations for "deterministically" selecting events for analysis appears to be arbitrary and contrary to the spirit of NRC's existing safety goals.

As we discussed in our recent comments on the Advance Notice of Proposed Rulemaking concerning severe accident requirements, we think that assumptions and acceptance criteria should be established for severe accidents, including event cutoff frequencies and consequence acceptance limits. Attachment 2 provides a copy of the relevant AECLT comments on the ANPR, Comment #4 and Comment #8.

B. SOURCE TERM

ISSUE: Should mechanistic source terms be developed in order to evaluate the advanced reactor and CANDU 3 designs?

AECLT COMMENT: The NRC Staff position, as stated on page 7, is that:

> "The CANDU 3 will also be different from LWR designs in certain respects. The coolant contains significant amounts of tritium. Following failure of a pressure tube, there is no heavy-walled reactor vessel to contain releases (there are large volumes of water in two concentric low-pressure tanks; moderator and shield water). Consequently the timing of releases is expected to be different from LWR's. Therefore, CANDU 3 also warrants a separate evaluation of source terms."

NRC Staff then recommends that the CANDU 3 source terms used for preapplication review be based upon mechanistic analyses and recommends specific guidelines for performing these analyses.

AECLT does not object to the NRC approach to evaluate our proposed source term during the preapplication review. We have a specific comment on the guidelines which is noted below. However, we do not agree that a pressurized heavy water reactor such as CANDU 3 is so fundamentally different from LWR's that it should require a different methodology for establishing source terms than that which NRC is now in the process of establishing for evolutionary LWRs. Specifically, the NRC Staff notes on page 7 of its draft letter that NRC Staff is now in the process of "...developing for LWR's a revision to the TID-14844 source term (NUREG-1465, "Accident Source Terms for Light Water Nuclear Power Plants," draft report for comment, June 1992)." AECLT requests that NRC give consideration to applying the same methodology which is developed for Advanced LWRs to Heavy Water Reactors during design certification reviews. In this way, the CANDU 3 will be judged on the same basis as the other water reactor designs currently under licensing review by the NRC.

In support of its position, AECLT notes that differences in design between PWRs and BWRs (for example the absence of a steam generator in BWRs) must be accounted for in source term calculations and the methodology being developed by NRC must allow for these differences in design. We believe that the differences in design between CANDUs and PWRs, insofar as they affect fission product transport after an accident, are of a similar nature. We note that CANDU's use the same type of zircaloy clad uranium oxide fuel as LWRs; hence the fuel behavior aspects of source term calculations should be the same. Also, the fact that tritium levels are higher in CANDU reactors because of the use of heavy water can be easily accounted for in the source term methodology.

We recognize that NRC has not completed its development and implementation of new source term standards for LWRs and therefore we agree with the approach recommended by the Staff for the preapplication review. However, we request that NRC approach the design certification review using the new standards now being developed by NRC for application to evolutionary LWRs.

With respect to the specific guidelines for development of mechanistic CANDU 3 source terms for the preapplication review, we believe the meaning of the term "credible severe accident" is unknown; we request that NRC clarify this term.

C. CONTAINMENT

ISSUE: Should advanced reactor designs be allowed to employ alternative approaches to traditional "essentially leak-tight" containment structures to provide for the control of fission product release to the environment?

AECLT COMMENT: The CANDU 3 design utilizes a traditional dry containment very similar to those in use on licensed LWRs and proposed for advanced LWRs. Therefore, AECLT does not believe the issue, as stated by NRC, is relevant to CANDU 3.

However, in the discussion section NRC states that "...for evolutionary LWRs, the Staff, in SECY-90-016, proposed to use a conditional containment failure probability (CCFP) or deterministic containment performance goal to ensure a balance between accident prevention and consequence mitigation. During the evolutionary LWR reviews, a great deal of careful review was necessary to assure that a probabilistic CCFP would not be used in a way that could detract from a balanced approach of severe accident prevention and consequence mitigation. For advanced designs and the CANDU 3, limited experience exists in the analysis and evaluation of severe accidents which could lead to significant difficulty and uncertainty in assessing a CCFP. For this reason, the Staff recommends that a deterministic containment performance goal be adopted for the CANDU 3."

We are concerned that NRC will evaluate CANDU 3 using different criteria than the criteria being used to evaluate containment of evolutionary LWR reactor designs. We do not see a good reason for this, and request that NRC consider instead evaluating the CANDU 3 containment using the same approach and criteria used in evaluating the containment designs for evolutionary LWRs.

With regard to the specific accidents to be used by NRC to evaluate containment performance, we have the following comment:

> "The Staff proposes to postulate a core damage accident as a containment challenge event and require that containment integrity is maintained for a period of approximately 24 hours after the onset of core damage."

We believe that a more specific definition of a "core damage accident" is needed. In this regard, we note that three types of events can lead to "core damage accidents": (1) reactivity events; (2) loss of heat sink events at high pressure; and (3) loss of heat sink events at low pressure. For the CANDU 3 design, the event frequencies of types (1) and (2) events will be less than 10^{-7} – low enough to be able to be considered "incredible." Thus, for the CANDU 3 design events of these types should not have to be considered when evaluating challenges to containment. Type 3 events comprise the "core damage accident" used as the "containment challenge" for the CANDU 3 design.

F. OPERATOR STAFFING AND FUNCTION

ISSUE: Should advanced reactor designs be allowed to operate with a staffing complement that is less than that currently required by LWR regulations?

AECLT COMMENTS: The draft Policy Issues paper states:

> "The CANDU 3 preapplicant has not proposed a specific number of licensed operators, but the Staff's expectation is that CANDU 3 will meet the current LWR staffing requirements."

AECLT does not understand why this issue was addressed to the CANDU 3 in the issues matrix but not in the text.

H. POSITIVE VOID REACTIVITY COEFFICIENT

ISSUE: Should a design in which the overall inherent reactivity tends to increase under specific conditions or accidents be acceptable?

AECLT COMMENT: AECLT agrees with the Staff's conclusion that "a positive void coefficient should not necessarily disqualify a reactor design." All reactors are subject to the insertion of positive reactivity under certain transient or accident conditions. The specific transient or accident varies with the reactor type. For the CANDU 3 the total void worth is between \$2 and \$3. While all such insertions raise significant concerns, the key question is whether the reactivity shutdown systems are reliable enough to reduce the frequency of reactivity insertions with a failure of all reactivity shutdown systems to an extremely low value (i.e., 10^{7} to 10^{10} per year).

In order to evaluate positive reactivity insertion in the CANDU 3 design, the Staff proposes to:

> "[require the analysis of] the consequences of events (such as ATWS, unscrammed LOCAs, delayed scrams, and transients affecting reactivity control) that could lead to core damage as a result of the positive void coefficients."

AECLT has several observations regarding this recommendation; including, concerning the meaning of the terms "ATWS" and "unscrammed LOCAs" as the terms might be applied to the CANDU 3. The terms were developed in the early days of LWR reviews and have specific meaning and special significance for those reviews.

1. ATWS: If the ATWS definition is retained with respect to the CANDU 3, its significance is diminished because of CANDU 3's multiple shutdown systems; namely, the Group 1 Regulating System and the two independent, diverse and redundant Group 2 Shutdown Systems.

   For a PWR, Anticipated Transient Without Scram (ATWS) is a faulted response to an anticipated initiating event requiring control element assemblies (CEA's) insertion for reactivity control. The initiating event is defined to be the occurrence of a transient requiring reactor trip for reactivity control coupled with failure of a trip to occur due to either mechanical failure of the CEAs to insert or the failure of both the Reactor Protection System (RPS) and the Alternate Protection System (APS) to generate a trip signal.

   Although 10 CFR 50.62 defined a prescriptive solution for the ATWS scenario in terms of prevention and mitigation, the success criteria for the event is given in NUREG-0460, Volume 3.

   For the limiting ATWS scenario, the criteria relating to the pressure boundary integrity and functionality of the valves required for long term cooling are of primary interest. The concern is that if the peak pressure in the RCS exceeds Level C stress limits (approximately 3200 psia), a breach of the primary coolant pressure boundary will occur and the Safety Injection System check valves will be jammed closed. This would result in a LOCA with no RCS makeup available.

Using the same definition for CANDU 3, an ATWS would be a faulted response to an anticipated initiating event requiring the Group 1 Mechanical Control Absorber rods (MCA) to be inserted for reactivity control. If the MCAs fail to insert, either of the two Group 2 shutdown systems remain poised to shutdown the reactor without a severe pressure transient.

2. Unscrammed LOCA:

For an "unscrammed LOCA", defined as a large LOCA requiring the insertion of the shutdown rods of the Group 2 Shutdown System (SDS1) for reactivity control, the Group 2 Shutdown System (SDS2) will insert poison into the moderator and will shutdown the reactor without resulting in a severe pressure transient.

3. Severe Accident End State Producing Positive Reactivity Insertion: For the CANDU 3 design, the severe accident End State producing positive reactivity insertion is shown by the CANDU 3 Accident Analysis to be a Failure to Shutdown when reactor shutdown by the Group 1 Regulating System and the two Group 2 Shutdown Systems has failed to occur. Consequences could include a mismatch between power production and the heat sink, resulting in severe fuel overheating and core damage. As discussed above in our accident evaluation comments, the CPSA gives an End State Frequency of a large LOCA with failure to shutdown of 10^{10} per year.

Acceptance criteria for severe core damage End States have not yet been established. The Staff proposes to take the frequency of positive reactivity insertion events into account in analyzing the phenomenon in the CANDU 3. Specifically,

> "The Staff's review of these analyses will take into account the overall risk perspective of the designs. [A requirement to change designs] will depend on the estimated probability of the accidents as well as the severity of the consequences."

AECLT agrees that consideration of the significance of positive reactivity insertion events should take into account the overall risk perspective of the designs. AECLT notes that acceptance criteria for such events have not yet been established. AECLT recommends that they be established during the preapplication review of the CANDU 3 design. In this regard, in its recent comments on the Severe Accident ANPR, AECLT provided its views on a selection process for severe accident events. See Attachment 2.

CATEGORY 2 CLASSIFICATION

The draft Policy Issues Paper identifies two issues for which the Staff recommends no departure from current regulations; namely, the Control Room Design and SSC Safety Classification issues. AECLT notes with concern that implementation of this recommendation will arbitrarily cut off review of new and innovative design approaches in these areas. AECLT asks that this recommendation be reconsidered. We believe that safety principles should govern and that new designs should be allowed to demonstrate how they meet such safety principles. Following such examination, the adequacy basis can be developed.

CONTROL ROOM AND REMOTE SHUTDOWN AREA DESIGN

ISSUE: Can current requirements for a seismic Category I/Class 1E control room and alternate shutdown panel be fulfilled by a Remote Shutdown Area, and a non-seismic Category I, non-Class 1E control room?

AECLT COMMENT: AECLT does not believe that the draft Policy Issues Paper accurately characterizes AECLT's approach to control room and Secondary Control Area (SCA) design. The draft paper states:

> "The main control room is not designed to be operable following an earthquake, tornado, fire, or loss of Group 1 (non-essential) electric power, but the operator must remain available to proceed to the secondary control area."

This statement is inaccurate. AECLT's approach to control room and SCA design is summarized in the points which follow. We request that the draft paper be revised in accordance with these points.

1. A CANDU 3 plant does not employ a "remote shutdown area" of the type connoted in the present NRC regulations and incorporated in current U.S. reactors. The CANDU 3 has a secondary control area (SCA) that is, in fact, a second control room. The SCA duplicates the control consoles available in the MCR for the control and monitoring of the Group 2 systems. The design basis for the man-machine interface in the SCA is a duplication to the fullest extent practical of control locations, layouts, and capabilities present in the MCR. The plant design basis requires that plant operators remain in the MCR if it is available and functional. The MCR is used to operate the plant safely under normal conditions and <u>most</u> accident conditions. Sufficient control and instrumentation are provided in both areas to shutdown the plant, achieve cold shutdown conditions, and maintain it in a safe condition under accident conditions including Loss-of-Coolant-Accidents. However, the MCR is designed for the effects of earthquakes and tornadoes to the extent of providing the operating Staff with protection from physical harm. Should the MCR become uninhabitable, control of the plant would be shifted to the SCA. The plant is designed such that all actions required to be accomplished while the plant operators shift control to the SCA are accomplished automatically. The route from the MCR to the SCA is qualified to allow its use in the event of earthquakes or tornadoes.

2. Regarding operability following a fire, AECLT wants to emphasize that no control room will remain operable following the control room fire required to be postulated by NRC fire protection requirements. However, the CANDU 3 design, which separates the plant into Group 1 and Group 2 areas, provides a significantly improved capacity to respond to fires; in that, a fire in any Group 1 area (including the MCR) will not prevent safe shutdown using Group 2 systems from the SCA and, likewise, a fire in any Group 2 area will not prevent shutdown from the MCR using Group 1 systems.

3. It is incorrect to characterize Group 1 systems as "non-essential" because it implies that Group 1 systems are "non-safety-related". The CANDU 3 design applies a graded level of design standards commensurate with the safety function to be performed in contrast to U.S. practice which applies extensive, safety-grade requirements to structures and systems that are safety-related and few, if any, requirements to those that are non-safety-related. The NRC Staff's statement in the draft SECY papers further implies that Group 1 power would be lost given a loss of offsite power. This is not correct. There are two redundant Group 1 diesel generator sets (as well as two redundant Group 2 diesel generator sets).

4. The separation of the CANDU 3 plant into Group 1 and Group 2 areas provides an enhanced capability to respond to other hazards that could render any control room inoperable. These hazards include sabotage, aircraft crashes, externally-generated missiles, smoke, and toxic gas. The CANDU 3 design also provides enhanced emergency planning capability by providing a redundant area for monitoring and control of essential plant parameters throughout all plant conditions from normal operation to cold shutdown.

In the draft Policy Issues Paper, the Staff discusses its reasons for recommending no departure from current regulations regarding control room design.

AECLT believes that the NRC Staff's evaluation approach to this issue appears to be more prescriptive regarding control room design than we believe is required by GDC-2 and GDC-17. The GDC permit a graded application of standards to structures, systems, and components commensurate with, as GDC-1 states, the importance of the safety functions to be performed. The importance of a control room in a plant that has essentially two control rooms is diminished from that in a plant with only one control room and a remote shutdown panel. Nevertheless, in either plant, the necessary functions need to be identified and the appropriate standards applied. The crux of this issue is the control room design envisioned in GDC-19. The CANDU 3 design vis-a-vis GDC-19 is discussed as follows.

The NRC Staff states that it is reluctant to approve any design that would increase the frequency of evacuation of the control room during design basis accident conditions or hamper the control or monitoring of upset conditions as the event progresses. AECLT is in general agreement with the NRC Staff position and believes that the CANDU 3 design satisfies the objective except for the low probability seismic and tornado events. As discussed above, AECLT feels the CANDU 3 design adequately addresses the concerns identified by the Staff regarding this issue and provides benefit to public health and safety. AECLT requests that the Staff reconsider its no departure recommendation.

ATTACHMENT 2

Excerpts from AECLT Comments on
NRC Advance Notice of Proposed Rulemaking
Concerning Acceptability of Plant Performance for Severe Accidents

[ref. AECLT letter to NRC dated December 21, 1992]

General Comment No. 4

4. Specifically, in the rule and implementing guidance the following matters should be addressed:

A. Selection Process for Severe Event Sequences Considered in the Design. The selection process should be based on event frequency. The process would establish the frequency limits to: (1)define the events requiring design changes to reduce their frequency, (2) define the events that require features to mitigate the event's consequences and (3) define events that need not be considered in the design.

B. Consequence Limits: For each event sequence defined by A(1) and A(2) above (e.g. reactivity events, loss of heat sink at High/Low Pressure), acceptable consequences for the event frequency should be defined on an overall basis (e.g. containment stress and leakage, radiological consequence limits). In addition, a phenomenon acceptance criterion should define the acceptable consequences for each individual phenomenon (e.g. hydrogen, molten fuel, non-condensable gas) associated with the event consistent with the overall acceptance criteria and the design features that produce the phenomenon.

C. Phenomenon Acceptance Criteria: For each phenomenon acceptance criterion, systems/features should be identified which provide the means to mitigate the consequences of the phenomenon.

D. System/Feature Design Criteria: For each system/feature, design criteria should be established for capacity, load combinations, environmental conditions vs time, and reliability. The reliability criteria should include: redundancy, diversity, power supply, separation (from each other and from systems/features whose failures are involved in the severe accident event sequences), and environmental qualifications.

E. System/Feature Demonstration Requirements: For each system/feature, the demonstration analysis/test requirements should be defined. These should include assumptions, acceptance criteria, analytical methods, and test requirements.

General Comment No. 8

8. As discussed in 3 and 4 above, a severe accident rule should specify a cut-off event frequency such that events below this frequency need not be considered in the design and for which further analysis is not required.

NUREG/CR-5368, "Reactivity Accidents" reported the results of analyses of light water reactor reactivity events performed by Brookhaven National Laboratory. For that effort, Brookhaven categorized potential event sequences as being worthy of further analysis, or not. One of the screening criteria used to determine the importance of a sequence for further analysis was whether the sequence required too many low probability events to occur in combination. Brookhaven established a screening methodology with which low probability events could be eliminated from further consideration.

Event sequences with a frequency of less than 1E-7 per reactor year were considered "incredible" and not recommended for further study.

AECLT believes that the generic severe accident rule should codify similar screening criteria.

ASEA BROWN BOVERI

January 25, 1993
LD-93-007

Project No. 680

Mr. Dennis M. Crutchfield, Associate Director
Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, D.C. 20555

Subject: Comments on Advanced Reactor Policy Issue Recommendations

Dear Mr. Crutchfield:

As requested, enclosed are the combined comments of ABB Combustion Engineering Nuclear Power and ABB Atom (collectively ABB) on the Nuclear Regulatory Commission's (NRC) draft report "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements". We are providing comments which we believe will further clarify the recommendations, more firmly establish their basis and update quoted PIUS design information.

ABB would also like to take this opportunity to set forth its position with respect to ultimate pursuit of Design Certification for advanced reactor designs, in this case PIUS. We undertake such a program because we believe there is a market for the advantages which could be derived from an advanced product that takes a fundamentally different approach to reactor safety in a decisive manner. Clearly, the primary advantages for potential licensees lie in reduced operating costs. Those operating cost reductions would come about primarily through amendment of existing regulatory requirements, which we believe are justifiable based on the improved safety performance of advanced reactor designs.

We are heartened to note that the NRC's proposed policy positions recognize the need for regulatory flexibility while preserving their charter to assure the public health and safety. We believe, however, that the NRC can go further than current proposals without degrading their role as regulator. The ABB comments provided in the Enclosure to this letter are geared toward the types of actions we believe are necessary and appropriate for us to continue to justify the expenditure of corporate resources in pursuit of advanced reactor designs.

ABB Combustion Engineering Nuclear Power

Combustion Engineering Inc | 1000 Prospect Hill Road, Post Office Box 500 | Telephone (20: | Fax (203) 285-

ATTACHMENT 2

If you have any questions, please do not hesitate to call me or Chuck Molnar of my staff at (203) 285-5205.

Very truly yours,

COMBUSTION ENGINEERING, INC.

C. B. Brinkman

C. B. Brinkman, Acting Director
Nuclear Systems Licensing

CBB:cmm
Enclosures: As stated

xc: T. Cox (NRC)
D. Scaletti (NRC)

Comments on Draft Policy Issues Analysis and Recommendations

General Comments

In the Recommendations section of the draft policy report, the NRC staff recommends that the Commission not proceed with generic rulemaking on any of the identified policy issues, at this time. ABB believes that this approach should be reconsidered. Rulemaking should not be postponed based on the amount of design detail available for the various advanced reactor designs and CANDU 3. The NRC could set appropriate safety acceptance criteria irrespective of the level of design detail submitted thus far. Moreover, while advanced designs may contain increased design and/or ultimate performance uncertainty, that is a matter for the scrutiny of the review process and not for overly restrictive acceptance criteria. If this is not the case, a clear explanation of the basis for promulgating more restrictive acceptance criteria should be provided. Having definitive regulations in place will play a large part in a vendor's commercial decisions to move forward with Design Certification and utility decisions to purchase and apply for an operating license. If advanced reactor designs, with their higher safety reliability, lower core damage risks and off-site release goals, cannot obtain relief from certain existing regulatory requirements, the motivation to move through the design certification and operating licensing processes may not be strong enough. A clearly defined playing field is of utmost importance to the continued progression of advanced designs.

ABB acknowledges that the generic rulemaking approach recommended above differs with the direction being pursued for evolutionary ALWR designs. We believe, however, that this is warranted since the evolutionary ALWR designs are already fully emersed in the design certification process and generic rulemaking could create schedular delays or differences with the design certification rulemaking. The advanced designs, however, are only at the preapplication review stage and thus can more fully benefit from rulemaking.

Policy Issue Comments

Accident Evaluation

It is imperative that the role of severe accidents be clear as applied to advanced reactor designs. This is an area where relief is sought from current regulatory requirements. At this time, no credible core damage sequences have been identified for the PIUS design. If this holds true throughout the review process, would PIUS still be required to consider the effects of progressively more unlikely events just to arrive at a hypothetical severe accident scenario and evaluate consequences?

What are the implications once those consequences are known? Even if the consequences turned out to be extremely severe, what would it mean if the event frequency were determined to be extremely incredible (including uncertainties)? ABB recommends that the NRC establish a frequency threshold below which a fault sequence could be dismissed (i.e., incredible, not meaningful to consider). Of course it would be up to the vendor to convince the NRC reviewers of the validity of their frequency evaluations and the uncertainties applied. In other words, exactly what constitutes the plant's design basis must be definitive.

Source Term

As mentioned above, at this time no credible core damage sequences have been identified for the PIUS design. As such, a source term based on some degree of core damage seems to represent an extreme penalty for PIUS. Currently, the only postulated source of radioactive contaminants is leaking fuel pins (not associated with accident scenarios). We recommend that the last sentence of the PIUS paragraph in the Preapplicants' Approach section be deleted. It is premature to indicate that PIUS will likely utilize the revised source term resulting from the NRC/industry effort. ABB recommends that The policy statement should provide for a mechanistically derived source term commensurate with the degree of justifiable core damage.

Containment

This policy issue is tied to the definition of accidents to be evaluated as well as the source term to be employed. Here also the question of the treatment of severe accidents in relation to design requirements is ambiguous. The statement of the policy issue seems to assume, a priori, that there will be fission product release to the environment resulting from core damage. Since no core damage is predicted for PIUS, no consequential release is predicted. If an advanced plant can sustain the validity of such a prediction under the scrutiny of the review process, relief from selected containment design requirements currently in the regulations or proposed by the draft policy paper must be available.

Specifically, design certification should not be contingent upon meeting on-site and off-site radionuclide release acceptance criteria that utilize artificially calculated releases based on an arbitrary core damage assumption. In other words, for this issue the second recommendation of the staff should not force unreasonable core damage scenarios. Satisfaction of the first recommended criteria, concerning on-site and off-site release acceptance criteria, is all that is necessary coupled with probabilistically plausible accident scenarios.

The description of the PIUS containment design presented in the draft policy paper is incomplete. The below grade containment structure is more accurately described as follows:

> "Below grade, the bottom part of the containment structure and the monolithic prestressed concrete vessel (which contains the reactor pool water and forms the lower portion of the reactor pressure boundary) are joined together by means of vertical prestressing tendons that are run up to the top of the PCRV proper. The steel liner on the containment inside, which ensures leak tightness, is a continuous liner covering the whole bottom area, the cylindrical walls, as well as the upper parts."

Emergency Planning

Another area where vendors seek to gain some degree of regulatory relief by recognition of the improved safety effectiveness afforded by advanced reactor designs is in the requirement for Emergency Planning. ABB believes that the opening sentences of the Recommendation section could be modified to more clearly reflect no need for immediate regulatory action on this issue. There is no need for a recommendation that "... licensees be required to develop off-site emergency plans." As noted in the paper, this requirement is already established in NRC's regulations. An alternate opening might indicate that,

> "The staff proposes no changes to the existing regulations governing Emergency Planning at this time. While it is desirable to establish the potential for and magnitude of off-site releases early on, there is no need to burden the preapplication review process by over focusing on Emergency Planning. Emergency Planning issues are more appropriately addressed during the Combined Operating License phase, when a specific utility and plant site would be identified. Regulatory direction should, however, be available at the start of the Design Certification phase so that any Emergency Planning implications on design can be addressed. Moreover, addressing Emergency Planning later in the licensing timeline allows for consideration of input to be developed by the ongoing NRC/industry dialogue on this subject, as well as resolution of the related Accident Evaluation, Source Term and Containment policy issues."

The policy paper notes that ABB did not explicitly communicate it's goals regarding emergency planning. ABB's goals are in line with those cited for the other advanced reactor designs. That is, early notification requirements could be reduced, detailed evacuation planning could be eliminated (or at least reduced), EPZ size could be reduced and offsite emergency exercises would be unnecessary. The request for regulatory relief is based on;

- satisfaction of the Environmental Protection Agency's lower level Protective Action Guidelines for dose,
- PIUS passive safety features which preclude core damage, and
- PIUS design features which allow substantially longer response times before intervention (to protect the public health and safety) would be required.

By no means does ABB imply that emergency planning can be disregarded altogether. Rather, ABB believes that emergency planning can be accomplished without involvement of significant offsite resources and without the need to implement emergency plans in the very short term following an incident. Clearly, a plan of action to deal with emergencies is a necessary part of safety and the defense-in-depth philosophy.

Reactivity Control System

Under the section Preapplicant's Position, the description of the safety grade reactivity control system should be updated to reflect the recent design change to inject boron via a Scram Valve System instead of by inflow through the lower density lock initiated via trip of a reactor coolant pump. By this change, plant shutdown is not precluded by blockage of the density lock(s). Boron inflow from the reactor pool via the lower density lock is the ultimate passive scram mechanism because it requires no system or operator action to initiate the safety function.

Additionally, the draft paper quotes the diameter of the density lock tubes as 3". This value is incorrect, the density lock tubes are of a nominal 57mm (2.25") diameter.

Operator Staffing and Function

ABB concurs with the staff position that operator staffing in advanced reactor designs could be reduced but that the appropriate staffing complement for a given design should be based on function and task analysis. The ability to shutdown with eventual progression to a safe shutdown condition following loss of computer control capability is an ongoing issue for current ALWR design certification applicants. ABB believes that the same guidance emerging for I&C diversity could be applied to PIUS which is also an ALWR.

Further, the recommendation to demonstrate adequacy on an "...actual control room prototype" should not be stated in absolute terms. Allowance should be permitted for consideration of applicable operating experience in place of a plant specific prototype. This modification would accommodate the body of experience from the new generation of digital computer controls being developed in conjunction with evolutionary plant designs and for non-nuclear applications. Such designs will likely see service before the proposed advanced reactor designs providing an experiential data base precluding the need for an actual advanced plant control room prototype.

Residual Heat Removal

No comment.

Positive Void Reactivity Coefficient

Although a summary table is included in the paper, it would be beneficial to indicate the applicability of specific issues to each of the designs under review in each policy issue write-up. For this specific issue it should be noted that positive void reactivity coefficient concerns are not applicable to the PIUS design.

Control Room and Remote Shutdown Area Design

Under the Preapplicant's Position section, the statement indicating manual scram for PIUS via trip of the main reactor coolant pumps should be updated to reflect the recent design change to a Scram Valve System as the primary (automatic) trip mechanism. Trip of the reactor coolant pumps remains available to the operator as a manual scram action. Boron inflow from the reactor pool via the lower density lock is the ultimate passive scram mechanism because it requires no system or operator action to initiate the safety function.

Safety Classification of Structures, Systems, and Components

No Comment.

Project No. 674

Department of Energy
Washington, DC 20585

JUN 2 1 1993

Mr. Stephen P. Sands
Project Manager, ALMR
Advanced Reactors Project Directorate
M/S 11-D-23
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Sands:

Subject: Commission Papers On Policy Issues And Schedules Concerning The Preapplication Reviews Of Advanced Reactor And CANDU 3 Designs

We received the subject documents, namely (1) a draft paper providing the staff's positions on ten policy issues and (2) a final paper, SECY-92-393, "Updated Plans and Schedules for Preapplication Reviews of Advanced Reactor (MHTGR, PRISM, and PIUS) and CANDU 3 Designs," and the staff's proposed schedules for the preapplication reviews. We appreciate the opportunity to comment on these documents and trust that our comments will be taken into consideration as the U.S. Nuclear Regulatory Commission (NRC) postulates its final position. The following comments pertain to the Advanced Liquid Metal Reactor (ALMR)-Power Reactor Innovative Safe Module (PRISM) design.

<u>Policy Issues</u>

The draft policy issues paper identifies ten issues as follows:

A. Accident Evaluation
B. Source Term
C. Containment Performance
D. Emergency Planning
E. Reactivity Control
F. Operator Staffing
G. Residual Heat Removal
H. Positive Void Reactivity
I. Control Room Design
J. Safety Classification

The staff states that eight of these issues apply to PRISM/ALMR. They further state that they believe departure from current regulations should be considered for seven of these issues while

ATTACHMENT 3

departure from current regulations is not warranted at this time for one issue. This categorization is presented below, with comments.

a. PRISM/ALMR issues for which departure from current regulations should be considered

A. Accident Evaluation
B. Source Term
C. Containment Performance
D. Emergency Planning
E. Operator Staffing
F. Residual Heat Removal
G. Positive Void Reactivity

The staff positions concerning these seven issues largely parallel positions taken previously by the PRISM/ALMR design team. A general characterization of these positions is that where deviations from current regulations are necessary, they should be performance based rather than prescription based, and that additional work is required to generate the data required to make final judgements. We agree with this approach.

However, the staff goes on to propose that "...where deviations are recommended, the staff proposes more conservative alternatives to the preapplicants' proposals to account for uncertainties associated with the conceptual design, which should ensure that conclusions made during the preapplication review will provide a reasonable basis for the detailed design being found acceptable at design certification."

Our concern with excessive conservatism in the preapplication review is that it will be difficult to reduce such conservatism as additional data are developed to support detail design, and that the end result may be alternative regulations serving as surrogates which create defacto new policy more conservative than the NRC safety goals.

To avoid this problem, we suggest that performance criteria should be developed which must be met regardless of uncertainties and design details. These performance criteria should be comparable to those of Light Water Reactors (LWR) and consistent with the safety goals. Allowance for uncertainties, commensurate with the level of knowledge and detail, should then be made in the evaluation process. As new information and details are developed, reevaluations should be made against the original performance criteria, using revised uncertainties appropriate for the level of new information and details.

b. PRISM/ALMR issues for which departure from current regulations is not warranted at this time

Control Room Design

The staff position for this issue differs from that taken by the PRISM/ALMR design team. Specifically, the staff recommends that until passive LWR policy for design requirements of control rooms and remote shutdown facilities is finalized, current LWR regulations and guidance should be applied to the preapplication review of advanced reactor designs. This would mean that safety-related equipment would have to be available to the operator in the control room, and that the control room/building would have to be designed and built to seismic Category 1 standards.

The PRISM/ALMR design team has presented the technical basis for the reference plant control system, reactor protection system, control room, and RSF design. This approach is performance based, and is believed to meet the general design criteria. It takes into account the passive and inherent safety features of the PRIMSM/ALMR design that go beyond those being considered in the passive LWR designs. We believe the staff should treat this issue similarly to the other seven issues by using a performance-based approach and allowing the PRISM/ALMR design team to develop the additional information required to demonstrate this performance.

We are ready to continue our discussions to resolve the outstanding issues without compromising the innovative design concept or impacting on the scheduled release of the Final Safety Evaluation Report.

Sincerely,

Nicholas Grossman

Nicholas Grossman
Director, Division of LMRs and Breeders
Office of Advanced Reactor Programs
Office of Nuclear Energy

cc:
James E. Quinn, GE
Richard Hardy, GE

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Department of Energy
Washington, DC 20585

January 27, 1993

Project No. 672

Dr. Thomas Murley, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Dr. Murley:

We have reviewed the advanced reactor review schedule information as presented in the Nuclear Regulatory Commission (NRC) document SECY-92-393 "Updated Plans and Schedules for the Preapplication Reviews of the Advanced Reactors Designs (MHTGR, PRISM, PIUS and CANDU 3)." We believe the Modular High Temperature Gas-Cooled Reactor (MHTGR) Preapplication Safety Evaluation Report (PSER) completion in December 1995 will not support the Department of Energy achieving the congressional intent contained in the Energy Policy Act of 1992 because the Department would need to submit the MHTGR preliminary design 9 months after receipt of the PSER.

As per our discussions, we believe a date of June 1994 for completion of the MHTGR PSER would be more appropriate. This date would be more in line with projections of SECY-91-161 "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions, May 31, 1991," and would mean that the MHTGR program could continue preliminary design and prepare the Preliminary Design Approval application with a clearer understanding of the NRC's view of MHTGR design issues.

Therefore, we request the NRC review resource allocations and consider a prioritization for the MHTGR preapplication review completion that will support a June 1994 final PSER.

Additionally, to ensure the NRC is fully aware of the Department's plan to implement the advanced reactor requirements of the Energy Policy Act of 1992, the Department will forward to the NRC a draft copy of the Advanced Reactor 5-year plan for their review and comment.

Sincerely,

E. C. Brolin

E. C. Brolin
Deputy Assistant Secretary
for Civilian Reactor Development
Office of Nuclear Energy

ATTACHMENT 4

Consumers Power
POWERING MICHIGAN'S PROGRESS
General Offices: 1945 West Parnall Road, Jackson, MI 49201 • (517) 788-0453

David P Hoffman
Vice President
Nuclear Operations

January 28, 1993

Mr. Dennis M. Crutchfield, Associate Director
Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: COMMISSION PAPERS ON POLICY ISSUES CONCERNING THE PREAPPLICATION REVIEWS OF ADVANCED REACTORS

Dear Mr. ~~Crutchfield~~ Denny:

As you know, for the past few years Consumers Power Company has actively participated in Modular High Temperature Gas-Cooled Reactor (MHTGR) development. We, and the other members of Gas-Cooled Reactor Associates (GCRA), provide the perspective of utilities that own and operate current nuclear plants. This utility role is unique among advanced reactor programs. A few weeks ago, the GCRA Management Committee received a briefing on the subject Commission papers. While we appreciate the diligent work of NRC staff in drafting those papers, we are concerned that they do not convey the higher vision of what is needed, and what might be achieved, from innovative reactor concepts, post Advanced Light Water Reactor (ALWR) deployment. We believe NRC policy guidance should be framed within benchmarks already established by the Commission, the realities of industry needs, and the related opportunities offered by advanced reactors; i.e.:

- The Advanced Reactor Policy Statement, "sets forth the general characteristics of advanced reactor design, which the Commission believes advanced reactors should exhibit, to increase assurance of safety, to improve public understanding, *and to promote more effective regulation.*" While current regulatory philosophy has been *effective* in protecting the public, there is mounting evidence that current regulatory processes fail to make *efficient* use of both industry and NRC resources. NRC policy guidance must recognize the need to make *effective* and *efficient* use of the Nation's resources, including those of the NRC, and foster the development of regulatory criteria and processes appropriate to advanced reactor technologies.

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BIG ROCK POINT
Nuclear Plant

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- NRC Safety Goals provide a framework for the consideration of risks from nuclear power plants. The nuclear industry has thus been provided with a threshold of acceptable risks to society based on several decades of experience with operating facilities. NRC policy guidance should build on this basis to delineate the risk-significance of plant features and personnel actions and, thereby, clarify the interface between the regulator and industry so that each may make productive use of resources.

- An Industrywide Initiative to reduce nuclear generation costs while assuring high levels of safety has been launched under the auspices of NUMARC. This effort grew out of Chairman Selin's acknowledgement that NRC may have contributed to O&M cost increases at nuclear plants by imposing regulations that do not necessarily increase safety, and his request for the industry to assist the Commission in identifying such regulations. It has been suggested by NUMARC president Joe Colvin that the current regulatory environment, the cultures of both the NRC and industry, have been molded by the accident at TMI. NRC policy guidance on advanced reactors must look beyond the current industry crises and be receptive to constructive changes in the regulatory environment and/or reactor technologies.

- The Energy Policy Act of 1992, Subtitle C, Advanced Nuclear Reactors, contains provisions for the ALWR and "...other advanced nuclear reactor technologies that may require prototype demonstration prior to commercial availability..." These provisions include, but are not limited to, high temperature gas-cooled reactors and liquid metal reactors. Provisions of the Act, offer the prospects of standardized, NRC certified commercial nuclear power plant designs, based on prototype tests. NRC policy guidance on advanced reactors should be receptive to the development of the appropriate regulatory framework for the reactor technology and should allow for the use of results from development programs, including prototype tests.

Given the Nation's investment in LWR technology, it follows that the LWR licensing framework is the starting point. However, the MHTGR (and possibly other advanced reactors) offers a diverse and contingent path to the resolution of technical issues underlying the business risks of owning a nuclear power plant. From the perspective of operating utility management, it may be inappropriate for NRC to strive for consistency with ALWR regulations as was frequently indicated in the subject Commission papers. Rather, policy guidance should provide for systematically identifying the systems, structures and components significant to public safety and the regulatory processes that should be applied to them, according to the attributes of the technology. In this regard, an Attachment to this letter elaborates our views on key policy issues for the MHTGR.

In summary, the NRC Advanced Reactor Policy Statement and Safety Goals should provide the framework for departure from current regulatory philosophy. Certainly the need for improved regulatory environment exists, as evidenced by the Industrywide Initiative directed at current plants. Finally, provisions of the Energy Policy Act provide a route for regulatory development in concert with reactor design and technology development. NRC policy guidance should be responsive to industry needs and receptive to the opportunities afforded by advanced reactors.

We look forward to providing continued support to this crucial area of MHTGR development.

Sincerely,

David P. Hoffman
Vice President of Nuclear Operations
Chairman, GCRA Management Committee

Copies with attachment:
Advisory Committee on Reactor Safeguards
GCRA Management Committee
GCRA Operations Working Group
MHTGR Program Participants

ATTACHMENT
GCRA PERSPECTIVE ON KEY MHTGR POLICY ISSUES

Containment Performance

The essence of the MHTGR safety concept is the capability to retain essentially all fission products within ceramic-coated fuel particles for the full range of licensing basis events. In this regard the MHTGR is fundamentally different than the ALWR. The MHTGR's capability derives from the attributes embodied in the design and manufacture of the fuel and reactor module. By simplifying reactor safety in this way, the MHTGR concept can potentially enhance the efficiency and effectiveness of nuclear safety regulation. Since the means to accomplish safety functions are embodied in the design, regulations may center on design verification and manufacture of the fuel and reactor modules, with a substantially reduced regulatory effort applied to site activities. In this regard, MHTGR demonstration tests will be of benefit to regulators, as well as to industry. This approach is inherently more efficient than a regulatory focus on site activities, since each plant involves a unique combination of owner, project organization, plant staff and public interest groups.

We are heartened by NRC staff's recognition of the potential to license a design with a containment concept tailored to the overall performance of the design. However, this position appears to be linked to an as yet undefined, arbitrary core damage assumption. From probabilistic risk assessments conducted to date, the MHTGR Program estimates that an event of sufficient severity so as to cause core damage would have a predicted frequency of occurrence less than 10^{-4} per plant year. The ACRS letter of October 13, 1988, noted that, "Neither the designers, the NRC staff, nor the members of the ACRS have been able to postulate accident scenarios of reasonable credibility, for which an additional physical barrier to release of fission products is required in order to provide adequate protection to the public." The MHTGR safety concept would essentially eliminate the potential for severe core damage and core disarray like the accident at TMI, an important consideration in gaining enhanced public acceptance of nuclear power.

Residual Heat Removal

The MHTGR accomplishes residual heat removal with systems, structures and components that are continuously functioning and do not require initiating signals, external power sources or moving parts. We note that NRC staff recognized the potential for relying on a single, completely passive, safety-related residual heat removal system. In addition, the MHTGR approach to residual heat removal is readily demonstrated and does not entail the potential for the interaction of passive and active systems. In this regard, the MHTGR is fundamentally different than the ALWR and, coupled with the approach to fission product retention described above, the MHTGR may allow a different approach to the regulatory treatment of non-safety related systems.

Safety Classification, Control Room Design, Emergency Planning

These issues are far too important to the commercial potential of the MHTGR concept (or any nuclear option) to be decided on the basis of judgment and existing precedent. It is here that the provisions of the Advanced Reactor Policy Statement (see especially the Commission Response to Question 1 regarding the use of less prescriptive licensing criteria) and the NRC Safety Goals must be confirmed by disciplined design and technology development programs. For the MHTGR to become commercially viable, regulations must be consistent with the risk-significance of plant equipment, personnel actions, and the provisions of emergency plans.

STAFF RESPONSE TO THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS) REVIEW OF POLICY ISSUES ANALYSIS AND RECOMMENDATIONS CONCERNING ADVANCED REACTORS

ENCLOSURE 5

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 24, 1993

Mr. Paul G. Shewmon, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Shewmon:

I am responding to the Advisory Committee on Reactor Safeguards (ACRS) letter of February 19, 1993, to the Chairman concerning the issues pertaining to the advanced reactor (PRISM, MHTGR, and PIUS) and CANDU 3 designs and their relationship to current regulatory requirements. The responses in the enclosure are organized into General and Specific categories, as are the comments in the ACRS letter. I have commented upon those issues where the staff's position is questioned or where I felt our position needed some clarification.

The staff will continue to keep the ACRS informed of any changes in staff or preapplicant positions, as well as any significant developments in the implementation of the policies.

Sincerely,

Original signed by
James M. Taylor

James M. Taylor
Executive Director
for Operations

Enclosure:
Staff Response to ACRS Comments

cc w/enclosure:
The Chairman
Commissioner Rogers
Commissioner Curtiss
Commissioner Remick
Commissioner de Planque
SECY

STAFF RESPONSE TO ACRS COMMENTS

GENERAL COMMENTS

Comment 1: We find that the identified issues are important and that the staff should receive guidance from the Commission. (There are other policy issues affecting these reactor designs that are being addressed in connection with the evolutionary and passive LWR designs.) There may well be additional policy issues that appear during the preapplication review process. The staff has committed to identify any such issues in subsequent Commission papers.

Response: The staff will identify any policy issues that arise subsequently.

Comment 2: The staff has grouped these ten issues into the two categories described above. We note that all of the affected preapplicants who appeared before us would treat Issue I (Control Room and Remote Shutdown Area Design) as a Category 1 issue, whereas the staff proposes it as a Category 2 issue. We will discuss this difference of opinion below in our opinion on Issue I.

Response: No response to this comment.

Comment 3: For Category 1 issues, the staff proposes more conservative alternatives than the preapplicants propose, in order to account for uncertainties associated with the conceptual design. We are concerned that such an approach might well freeze an unnecessarily large degree of conservatism into the designs, and the preapplicants would have great difficulty persuading the staff to relax this conservatism on the basis of more precise information available in the final design.

Response: The staff's positions on Category 1 issues, with the exception of containment and emergency planning (EP), are not more conservative than positions proposed by preapplicants. The containment recommendation is based on, and will be further developed from, the current Advanced Notice of Proposed Rulemaking issued by the Commission, and the staff's EP recommendation notes that further evaluation of the advanced reactor designs may permit some relaxation from current requirements.

The staff is not prepared to, nor does it intend to, specify firm standard design certification requirements at the preapplication review stage. The preapplication review approach is to examine a range of postulated operational events and accidents, including some accident sequences postulated to result in either substantial core damage alone or core damage with a large release from containment. Such examinations are consistent with the direction of the Commission's Severe Accident Policy Statement, and with the

10 CFR Part 52 requirement to confirm, through a full-plant, design-specific probabilistic risk assessment (PRA), the margins available in the design to accommodate events of low probability. These examinations will assist in specifying which accident sequences should be assigned to the frequency categories identified as anticipated operational events, anticipated transients and accidents, and severe accidents. It is intended that licensing basis consequence limits for accident sequences will be entirely consistent with the Commission's safety goal and severe accident policies.

For the preapplication review, the staff intends to develop a set of event categories for accident selection. These categories would be used for defining acceptable analyses criteria and consequences (dose limits) based on the likelihood and potential significance of an event.

These categories would both encompass traditional events at light-water reactors (LWRs) and would extend to severe accidents. For anticipated operational transients, analyses assumptions would be conservative. For anticipated or design base accidents, the single-failure criterion would be used in conjunction with such conservative analyses assumptions as no operator credit and no use of non-safety-related equipment to mitigate the outcome of an event. For severe accidents, a best-estimate analyses assumption would be acceptable as would justifiable operator actions and credit for non-safety-related equipment.

The analyses would, therefore, prescribe the event scenario, identifying the amount, timing, and magnitude of fuel and core damage, and the status of the integrity of the primary coolant boundary. This fuel and core damage would then be used to evaluate the amount and type of radionuclide releases from the fuel into the primary coolant. Uncertainties in the specific amount of radionuclide species released from the fuel would be assessed for the scenario but would not be further increased to provide some measure of additional conservatism. The status of the primary coolant boundary would then determine the releases from the primary coolant boundary into the containment (or confinement) region, and the containment performance would then determine the releases to the environment. With regard to conservatism, the effectiveness of systems and barriers to mitigate releases would be treated consistent with the appropriate event category.

In addition to using PRA results to identify an appropriate event category for some scenarios, the staff may, based on engineering judgment, require the preapplicant to evaluate some event sequences applying appropriate conservatism to, for example, the anticipated accident category rather than the severe accident category. The staff would need this assessment in order to assess margins in the design, to account for uncertainties in the PRA data base, and to ensure that the importance of systems needed to mitigate accidents is understood.

Comment 4: We support the staff recommendation that "a prototype CANDU 3 is not required for design certification."

Response: The staff and ACRS agree on the staff recommendation.

Comment 5: We support the staff intention to notify the Commission if its position on any of these ten issues should change, or if new issues are identified.

Response: The staff and ACRS agree on the staff recommendation.

Comment 6: We have no objection to the staff recommendation that the highest priority be given to issues that are applicable to the PRISM design.

Response: The staff and ACRS agree on the staff recommendation.

Comment 7: We understand and sympathize with the staff recommendation to defer decisions on generic rulemaking on these ten issues. Nevertheless, we urge the Commission to address these decisions in the near future. (The generic rulemaking question may arise in connection with passive LWR designs.)

Response: The staff and ACRS agree on the staff recommendation.

Comment 8: In several places in the draft Commission paper, there occurs qualitative language, e.g., "appropriate conservatisms" or "credible severe accidents." This language must ultimately be translated into quantitative guidance. We believe that the quantitative guidance is, to a large measure, policymaking, and should not be relegated to low-level reviewers.

Response: The staff work on the advanced designs has not progressed to where it can quantitatively define "appropriate conservatism" or "credible severe accidents." We agree that such definitions should be approved at appropriate agency management levels, and will not be relegated to "low-level reviewers." The staff will delete the words "appropriate" and "credible" from the policy issues paper, as suggested by the ACRS.

SPECIFIC COMMENTS

Category 1 Issues

Comment: A. Accident Evaluation

The staff proposal to develop a single approach with certain specified characteristics appears reasonable. We would like to review that approach when it is ready. We believe, however, that the staff should identify at an early stage quantitative guidelines and criteria for accident selection and evaluation. We note that AECLT has taken exception to some of the statements in the draft Commission paper that relate to its approach to this issue. We believe that this disagreement can be resolved by AECLT and the staff.

Response: The staff plans to propose, as soon as possible, quantitative guidelines for design certification accident selection and evaluation. This will include definitions of frequency categories and associated acceptable consequence limitations assigned to categories and specified accident sequences.

Comment: B. Source Term

The staff proposal to base the source terms on mechanistic analyses appears reasonable, although it is clear that the present data base will need to be expanded. We note that the staff is now developing for LWRs a revision to the TID-14844 source term. It will be appropriate for the staff to consider using the newer approach when it develops source terms, and to take specific account of the unique features of each of the reactor types.

Response: The staff and ACRS agree on the staff recommendation.

Comment: C. Containment

The staff proposal "to postulate a core damage accident as a containment challenge..." appears reasonable. We would like to review the list of postulated accidents when it is ready.

Response: The staff and ACRS agree on the staff recommendation.

Comment: D. Emergency Planning

The staff proposes that advanced reactor licensees be required to develop offsite emergency plans which will include a requirement for onsite and offsite exercises. This proposal appears reasonable under the present circumstances, except that we would follow existing LWR guidance that permits the omission of offsite exercises when it can be shown that the design would preclude any accidental release exceeding the EPA Protective Action Guides. The staff has agreed to consider, after a review of Accident Evaluation (Issue A, above), whether some relaxation from current requirements may be appropriate. We urge that work on Issue D be closely correlated with work on Issues A and B, in order to avoid unnecessary conservatism.

Response: The staff and ACRS agree on the staff recommendation.

Comment: E. Reactivity Control System

The staff proposal that the absence of control rods need not disqualify a reactor design, provided that an applicant can show a level of safety in reactor control equivalent to that of a traditional rodded system, appears reasonable. We note that this issue is applicable only to the PIUS concept, and that we have not yet had the benefit of presentations by the PIUS designers.

Response: The staff and ACRS agree on the staff recommendation.

Comment: F. Operator Staffing and Function

The staff intends to review the justification for a smaller crew size by evaluating the function and task analyses for normal operation and accident management. This intention appears reasonable, although we believe that particular attention needs to be given to multiple module designs. We note that this issue is related to a similar issue for passive reactors. We believe that the Commission policy should be the same for the advanced reactors and CANDU 3 as it is for the passive reactors.

Response: The staff and ACRS agree on the staff recommendation.

Comment: G. Residual Heat Removal

The staff belief that reliance on a single, completely passive, safety-related residual heat removal (RHR) system may be acceptable appears reasonable, although we would have liked to see the criteria to be used by the staff in deciding acceptability. We agree with the staff that NRC regulatory treatment of non-safety-related backup RHR systems for these reactors should be consistent with design requirements (not yet identified) for passive LWRs.

Response: The staff and ACRS agree on the staff recommendation.

Comment: H. Positive Void Reactivity Coefficient

We agree with the staff that the existence of a positive void reactivity coefficient is a significant concern, but that it should not necessarily disqualify a reactor design. The burden of showing that the consequences of those accidents that would be aggravated by a positive void reactivity coefficient are either acceptable or could be satisfactorily mitigated by other design features surely falls on the preapplicant. On the other hand, the staff should state the criteria it will use to judge "acceptable" or "satisfactorily."

Response: The staff and ACRS agree on the staff recommendation. The staff will define the criteria to judge "acceptable" and "satisfactorily" as it performs the preapplication review and during this review will develop an understanding of the behavior of the particular design during positive reactivity insertion transients and accidents.

Category 2 Issues

Comment: I. Control Room and Remote Shutdown Area Design

We do not agree with the staff decision to treat this issue as a Category 2 issue, and the concomitant recommendation to apply current LWR regulations and guidance until passive LWR policy in this area is finalized. We believe that this issue should be a Category 1 issue, and that the preapplicants should accept the burden of convincing the staff that a proposed design is satisfactory, according to some criteria that should be specified by the staff.

Response: The staff is recommending a policy position to be utilized at the preapplication review stage. As stated in response 3 (above), the staff does not intend to shut off further dialogue with preapplicants, but to state that, at this time, justification for recommending departure from current requirements is not established. As noted in the policy paper recommendation, this issue also is dependent on further development within the context of advanced passive plant design reviews. The staff believes that preapplication design review should be completed by identifying and evaluating differences between preapplicant proposals and current LWR requirements, before recommending design certification requirements.

The staff believes that its current recommendation in the policy paper remains appropriate, that is, to evaluate preapplication designs by comparison to current LWR requirements/guidance for the immediate future. As advanced passive LWR policy for design requirements of control rooms and remote shutdown facilities is developed, and as further evaluation of the completely passive shutdown and decay heat removal functions in the Department of Energy designs are further evaluated, additional changes to relax requirements on main control rooms may be justified for the designs treated in this paper.

Comment: J. Safety Classification of Systems, Structures, and Components (SSCs)

This issue is relevant only to the MHTGR concept. GA makes a persuasive case that the MHTGR is sufficiently different that the LWR criteria for identification of safety-related structures, systems, and components should not arbitrarily be applied to the MHTGR. We concur with this view and believe that Issue J should also be classified as a Category 1 issue. This would not preclude coordination of the policy for passive reactors with the policy for the MHTGR.

Response: The staff believes that this issue is fundamental to the maintenance of the "defense in depth" philosophy and policy held by the Commission over many years of LWR regulation. The Nuclear Regulatory Commission (NRC) has long held it important to protect the integrity of the reactor coolant pressure boundary, to protect the SSCs necessary to shut down the reactor and maintain decay heat removal, and to protect the SSCs needed to prevent or mitigate accidents and minimize releases.

This policy is believed sound and necessary to the partitioning of the risk of radioactive releases among several independent barriers to radionuclide transport. The approach taken by the preapplicant for the modular high-temperature gas-cooled reactor (MHTGR) would limit NRC regulation to requiring safety-grade classification only on SSCs needed for preventing or mitigating offsite releases. The staff feels that this approach could result in placing reliance on too few SSCs, therefore, not assuring multiple, highly reliable barriers to radionuclide release.

This issue deals only with determining which SSCs should be classified as safety related, making it equivalent to the specification of SSCs as "safety grade." For advanced reactor designs, the classification as safety grade may not necessarily mean that such SSC should be designed, built, tested, and maintained to a single set of standards that includes seismic Category I, electrical Class 1E, and other traditional "safety grade" attributes. That is, having specified which SSCs should carry the safety-grade classification, consideration should be given, in the language of General Design Criterion 1, to design, fabrication, erection, and testing to quality standards commensurate with the importance of the safety functions to be performed.

A.D. Hink

Vice-President and
General Manager

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Project 679

January 25, 1993

Mr. Dennis M. Crutchfield
Associate Director for Advanced
Reactors and License Renewal
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Re: Commission Papers on Policy Issues and Schedules Concerning the Preapplication Reviews of Advanced Reactors and CANDU 3 Designs.

Dear Mr. Crutchfield:

This letter is in response to your letter of December 16, 1992, which provided AECL Technologies (AECLT) with two NRC Staff papers concerning the preapplication review of the CANDU 3 design. One paper was SECY-92-393 concerning "Updated Plans and Schedules for the Preapplication Reviews of the Advanced Reactor (MHTGR, PRISM, and PIUS) and CANDU 3 Designs." The other was a draft SECY paper entitled "Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements." We have reviewed the two papers and address each in turn.

SECY-92-393 establishes a revised schedule for completion of the preapplication review of the CANDU 3 design. According to SECY-92-393, the draft Preapplication Safety Evaluation Report (PSER) is to be issued in June 1994 and a final PSER in December 1994. This represents a significant change of twelve months over the earlier scheduled completion of June 1993 for the draft PSER. The June 1993 date provided time for AECLT to address any issues raised in the PSER and to submit its application for certification of the CANDU 3 design in the early part of the 1995-96 timeframe. AECLT has chosen the 1995-96 timeframe because it would allow a certified CANDU 3 design to be available to U.S. utilities in the expected timeframe for the placement of new orders for nuclear power plants. The new schedule of December 1994 pushes submission of the CANDU 3 design certification

ATTACHMENT 1

application to the end of the 1995-96 timeframe, with little to no allowance for contingency or slippage. Therefore, it is critically important that the new schedule in SECY-92-393 be faithfully adhered to and not extended.

In providing the Commission with relevant background on the CANDU 3 design, SECY-92-393 indicates that Atomic Energy of Canada Limited (AECL) is "negotiating to start construction in a Canadian province which could serve as a prototype for the CANDU 3 design in the U.S." and that AECL "would re-evaluate its design certification plans in the U.S. if Canadian construction plans did not materialize." Similarly the draft issues paper states that "a CANDU 3 reference plant is a key element in [AECLT's] plan for standard design certification." These statements require clarification in two aspects.

First, as the draft issues paper makes clear, and AECLT fully endorses, the CANDU 3 design is an evolutionary heavy-water design deriving from CANDU designs operating in Canada and elsewhere, for which there is over 200 reactor years of full power operating experience. Consequently, a prototype CANDU 3 is not required for design certification. Also, as the draft issues paper makes clear, while a reference plant built in Canada would greatly benefit the Staff's review of the CANDU 3 design, building such a plant is not necessary for certification of the CANDU 3 design. Rather, what is of importance is the relevant operating experience of the CANDU plants from which the CANDU 3 design evolved.

Second, as to potential availability of a reference plant in Canada, AECLT is pleased to inform the NRC that on December 21, 1992, the Government of Saskatchewan and AECL signed a Memorandum of Understanding (MOU). The MOU provides, among other things, for completion of the design and engineering for the CANDU 3, including the contribution of $20 million in matching funds by the Government of Saskatchewan to those being contributed by AECL. These funds are in addition to the approximately $100 million already spent by AECL over the past 5 years. In this and other respects, the relationship between the Government of Saskatchewan and AECL is similar to that between the Department of Energy and the Advanced Reactor Corporation in the U.S. First-of-a-Kind-Engineering effort. The MOU represents further progress in the advancement of CANDU technology as embodied in the CANDU 3. This progress notwithstanding, it is important to understand that our intent to go forward in the 1995-96 timeframe with an application for certification of the CANDU 3 design in the United States is independent of any schedule for building a CANDU 3 reference plant in Canada.

The draft Policy Issues Paper discusses the present scope of the CANDU 3 pre-application review, indicating that the Staff has revised the scope of the issues considered at the preapplication review stage, limiting them "to those which could affect the licensability of the proposed design." AECLT is looking to the pre-application review to resolve all issues identified during the review, in the sense that the PSER identifies the information which

AECLT must provide in the CANDU 3 design certification application in order for the Staff to successfully complete its review of that application.

AECLT is especially pleased to have the opportunity to comment on the draft Policy Issues Paper prior to its being finalized for submission to the Commission. AECLT would like to address the six substantive issues identified in the draft paper as relating to the CANDU 3 design; specifically, Accident Evaluation, Source Term, Containment Performance, Operator Staffing, Positive Void Reactivity and Control Room Design. In formulating these comments, AECLT has followed the Staff's distinction between Advanced Reactor issues and CANDU 3 issues and, consequently, addressed only those comments specifically applying to the CANDU 3 design. These issues are discussed in detail in Attachment 1 to this letter.

If you have any questions regarding this letter or the attachment, please do not hesitate to call.

Very truly yours,

A. D. Hink

A. D. Hink
Vice President/General Manager
AECL Technologies

Attachments: As stated

cc: Janet Kennedy, NRC
CANDU 3 Project Manager

STAFF RESOLUTION OF COMMENTS ON DRAFT POLICY PAPER

NRC RESPONSE TO ATOMIC ENERGY OF CANADA, LIMITED, TECHNOLOGIES (AECLT) (See Attachment 1)

Policy Issue A — Accident Selection

(1) The staff did not intend to imply that the frequency of events "considered" was cut off at 10^{-6} per year, but that the information given states [on page 1-14 of the 1989 part of the conceptual safety report (CSR) Part 3, "Safety Analysis"], "As in previous CANDU probabilistic safety assessments, events with frequencies less than 10^{-6} per year are not considered to be of high enough frequency to require a consequence analysis." AECLT pointed out that for at least one case a consequence evaluation is performed. This is the large loss-of-coolant accident without emergency core cooling design basis event used to determine the containment design pressure. AECLT estimates the frequency of this event is 7.6×10^{-7} per year. The acceptance criteria used by AECLT, for this dual-failure event, are similar to NRC 10 CFR Part 100 guidelines as applicable to design-basis accidents. The staff added this point to the text in the policy paper.

In the CANDU 3 CSR submitted to the staff, AECLT identifies four event categories in Part 3, "Safety Analysis," for consideration in accident analysis. The first category, A, is used in the initial design phase to establish design requirements for safety systems. The acceptance criteria for category A events include consequence limits similar to current limits in 10 CFR Part 100. The second category, B, is based on probabilistic safety assessment and the "consequences analyses will use realistic assumptions" [on page 1-11 of the CSR Part 3]. The frequency of events in category B appear to range from 10^{-1} to 10^{-6} per year based on the page 1-14 of the CSR reference to Figure 1-1, "Acceptance Criteria for Probabilistic Safety Assessment Studies." In Section 6.2 of the April 1988 part of the CSR, in a document titled "CANDU 300 - Systematic Review of the Plant Design, Revision 0," AECLT states that the assessment of events in category B "requires that the resulting event sequence be developed in a realistic fashion, and that the analysis assumptions be best estimate or design centre." Two additional categories, C and D, are described for which an assessment of the design is performed. Category C events are evaluated to assess the plant's capability to safely shut down, remove decay heat, and isolate containment. Among category D events is the assessment of catastrophic failures of large components to assure that they can be excluded from the design and analysis basis of the plant. AECLT has only submitted limited information for category A analyses in the CANDU 3 safety analysis section of the CSR. AECLT lists 37 category B events and notes that 23 of these "have not been assessed as part of the CPSA (conceptual probabilistic safety assessment."

(2) The staff did not intend to imply that reactivity insertion events are excluded from "consideration" by AECLT, but that consequence analyses will be necessary for selected event sequences that lead to severe core damage, even though the estimated frequencies may be less than 10^{-6} per year. The staff revised the text in the policy paper to clarify this point.

(3) The second reported quotation on page 3 of AECLT's comments does not appear in Section 5.2 of Appendix C of the CSR but is found in Section 6.2 of the April 1988 document titled "CANDU 300 - Systematic Review of the Plant Design, Revision 0." Similar words do exist in the 1989 part of the CSR, as noted in item 1 (above), but the underlined text ("In those cases where they are, Figure 2 will be extrapolated as necessary.") does not appear there. Nor does AECLT describe how such a determination to consider events below 10^{-6} per year would be made or what evaluation would be performed. The NRC preliminary review to identify policy issues has focused on the actual safety analyses performed for the CANDU 3 design in the 1989 part of the document. The staff will resolve the inconsistencies in the preapplicant's documentation as the review proceeds.

(4) The staff agrees that the primary focus of the AECLT analyses is on design requirements and acceptance criteria for the design of safety systems. Therefore, the staff believes that the concerns regarding severe accident assessment, consistent with 10 CFR 52.47, are valid.

(5) The staff agrees that design-basis accidents need to be evaluated in order to design safety systems, regardless of the perceived frequency of their occurrence. These are deterministically based scenarios selected by the designer to establish performance requirements for safety systems. However, the staff is also concerned that there may be other events of similar, or perhaps higher, frequency than some of these events that might lead to unacceptable consequences. This results from AECLT's proposal to use best-estimate analyses for all events not grouped into category A as compared to our current practice of using conservative analysis assumptions for all design basis accidents which would fall within the category B range. These category B events appear to fall in the frequency range of 10^{-1} to 10^{-6} per year.

(6) As pointed out in the staff's response to comments 1 through 5 (above), the staff concludes that the AECLT approach seems inconsistent with the provisions of 10 CFR 52.47 with respect to performing detailed consequence analyses of severe accidents.

(7) The staff agrees to include CANDU 3 specifically by name in the overall approach. The staff revised the text in the policy paper accordingly.

(8) In the "recommendation" section of the policy paper, the staff states that "Methodologies and evaluation assumptions will be developed for analyzing each category of events consistent with existing LWR practices." The staff intends to apply "the historical requirement for

conservative analysis for design-basis accidents (DBAs) and best-estimate analyses for beyond DBAs." The staff notes that the AECLT assumption used for category B events may be inconsistent with this traditional approach. The staff's intent to use this traditional approach has been clarified in the text of the policy issue paper.

Policy Issue B — Source Term

The staff intends to apply the methodology currently being developed in NUREG-1465, "Accident Source Terms for Light Water Nuclear Power Plants" (a draft report issued for comment), to the CANDU 3 design to the extent possible. The staff did not intend to imply that a unique method of assessment was required for a pressurized heavy-water reactor design, but that direct use of current LWR source terms might not be appropriate considering the differences in the designs. The staff revised the text in the policy paper to clarify this point.

Policy Issue C — Containment

The staff believes this issue applies to CANDU 3 because of the high leakage allowable, 2 percent per day by test, as compared to the current criterion of 0.5 percent per day for current-generation LWRs. AECLT agrees that this issue does apply to the CANDU 3 design.

Policy Issue F — Operator Staffing and Function

The staff agrees with the AECLT position and has removed CANDU 3 from the applicability matrix.

Policy Issue H — Positive Void Reactivity Coefficient

The issue is whether or not the staff should accept a design in which the positive void reactivity increases dramatically during certain events. The staff intends to require that AECLT analyze the consequences of events which could lead to large reactivity insertions. This will include events involving failure to shut down.

Policy Issue I — Control Room and Remote Shutdown Area Design

The staff acknowledges AECLT concern with the issue, but at this time the staff's recommendation, as presented to the Commission for guidance, is to require a seismically and electrically qualified main control room.

NRC RESPONSE TO ASEA BROWN BOVERI-COMBUSTION ENGINEERING
(See Attachment 2)

Policy Issue A — Accident Evaluation

The staff believes that it is necessary to address severe accidents consistent with the provisions of 10 CFR 52.47 with respect to performing detailed consequence analyses of severe accidents. The staff has not established a frequency below which a fault sequence could be dismissed without further evaluation. As the preapplication review progresses, the staff plans to work with ASEA Brown Boveri-Combustion Engineering (ABB-CE) to ensure that the severe accident issue is addressed and that the likelihood and consequences of these accidents are understood.

Policy Issue B — Source Term

It is the staff's intent to apply the methodology currently being developed in NUREG-1465, "Accident Source Terms for Light Water Nuclear Power Plants" (a draft report issued for comment), to the PIUS design to the extent possible. The staff did not intend to imply that the LWR source terms developed for pressurized-water reactor designs would be used for PIUS, but that methodology to evaluate release of radionuclides from the fuel and the subsequent transport to the environment would be used to assess the design during the preapplication review. The staff revised the text in the policy paper to clarify this point.

Policy Issue C — Containment Performance

The staff has revised the technical description of the PIUS containment design in the text of the policy paper.

Policy Issue D — Emergency Planning

The staff has revised the text in the policy paper to describe the conditions under which ABB-CE would seek regulatory relief.

Policy Issue E — Reactivity Control System

The staff has revised the technical description of the PIUS scram system design in the text of the policy paper.

Policy Issue F — Operator Staffing and Function

The staff is currently recommending that an "actual control room prototype" be used for test and demonstration purposes. The staff, as part of its consideration for design certification, would consider any information provided by an applicant to determine if such information adequately addresses staff concerns.

Policy Issue H — Positive Void Reactivity Coefficient

The staff has revised the text of the policy paper to indicate that this issue does not pertain to the PIUS design.

Policy Issue I — Control Room and Remote Shutdown Area Design

The staff has revised the technical description of the PIUS scram system design in the text of the policy paper.

NRC RESPONSE TO THE UNITED STATES DEPARTMENT OF ENERGY (DOE)/PRISM DESIGN (See Attachment 3)

Policy Issue I — Control Room and Remote Shutdown Area Design

The staff acknowledges the U.S. DOE's concern with this issue for the PRISM design but, at this time, the staff's recommendation, as presented to the Commission for guidance, is to require a seismically and electrically qualified main control room.

NRC RESPONSE TO THE UNITED STATES DEPARTMENT OF ENERGY (DOE)/MHTGR DESIGN (See Attachment 4)

Policy Issue G — Residual Heat Removal

The staff did not intend to imply that the resolution of the treatment of non-safety-grade equipment resulting from efforts on the passive light-water reactor designs would be directly applied to MHTGR. Rather, the staff notes that these are ongoing activities and the staff intends to factor these studies into any recommendations concerning the advanced reactor design; also, as stated in the paper, the staff "will assure that NRC regulatory treatment of non-safety-related backup RHR systems is consistent with Commission decisions on passive light-water reactor design requirements."

Policy Issue I — Control Room and Remote Shutdown Area Design

The staff acknowledges DOE's concern with this issue for the MHTGR; but, at this time, the staff's recommendation, as presented to the Commission for guidance, is to require a seismically and electrically qualified main control room.

Policy Issue J — Safety Classification of Structures, Systems, and Components

The staff acknowledges DOE/MHTGR's concerns on safety classification however at this time the staff's recommendation, as presented to the Commission for guidance, is to maintain the current defense-in-depth philosophy in defining those structures, systems, and components important to safety.

NRC RESPONSE TO THE GAS-COOLED REACTOR ASSOCIATES (GCRA)
(See Attachment 5)

Policy Issue C — Containment Performance

The staff believes that it is necessary to address severe accidents consistent with the provisions of 10 CFR 52.47 with respect to performing detailed analysis of the consequences of severe accidents. The staff has not established a frequency below which a fault sequence could be dismissed from further evaluation. As the preapplication review progresses, the staff plans to work with the preapplicant to ensure that the severe accident issue is addressed and that the likelihood and consequences of these accidents are understood.

Policy Issue G — Residual Heat Removal

The staff did not intend to imply that the resolution of the treatment of non-safety-grade equipment resulting from efforts on the passive light-water reactor designs would be directly applied to the MHTGR. Rather, the staff notes that these are ongoing activities and the staff intends to factor these studies into any recommendations concerning the advanced reactor design. As stated in the paper, the staff "will assure that NRC regulatory treatment of non-safety-related backup RHR systems is consistent with Commission decisions on passive light-water reactor design requirements."

Policy issue I — Control Room Design and Remote Shutdown Area

The staff acknowledges GCRA's concern with the issue; but, at this time, the staff's recommendation, as presented to the Commission for guidance, is to require a seismically and electrically qualified main control room.

Policy Issue J — Safety Classification of Structures, Systems, and Components

The staff acknowledges GCRA's concerns about safety classification; however, at this time, the staff's recommendation, as presented to the Commission for guidance, is to maintain the current defense-in-depth philosophy in defining those structures, systems, and components important to safety.

A SUMMARY OF CURRENT DESIGNS FOR THREE
ADVANCED REACTORS AND CANDU 3

A. CANDU 3 (CANADIAN DEUTERIUM URANIUM 3) REACTOR

Development History

The CANDU 3 is the latest version of the pressurized heavy-water reactor (PHWR) system developed in Canada. The CANDU 3 design evolved from other CANDU PHWRs, most notably the CANDU 6 design. The CANDU 3 is a generic standard design that has retained many key components (e.g., steam generators, coolant pumps, pressure tubes, fuel, on-line refueling machines, and instrumentation) that have been proven in service on operating CANDU power reactors. Currently, there are 25 CANDU reactors in operation in 6 different countries and 19 under construction. The first CANDU reactor was placed in service in 1968. CANDU experience to date amounts to more than 175 years of effective full-power operation.

On May 25, 1989, Atomic Energy of Canada, Limited, Technologies (AECLT) informed the NRC of its intent to submit the CANDU 3 reactor design for standard design certification in accordance with 10 CFR Part 52. AECLT of Rockville, Maryland, is a wholly owned subsidiary of Atomic Energy of Canada, Limited (AECL) (a crown corporation of Canada), and is the preapplicant for the CANDU 3 design. AECL in Canada is also pursuing standard design certification of the CANDU 3 with the NRC's Canadian counterpart, the Atomic Energy Control Board of Canada. AECLT plans to submit a standard design certification application for CANDU 3 sometime in 1995 or 1996.

Design Description

The CANDU 3 is a 450-MWe, heavy-water-cooled and -moderated, horizontal pressure tube reactor that evolved from the CANDU 6 design. The CANDU 3 uses deuterium oxide (heavy water) as a moderator because its small thermal neutron capture cross-section allows the use of natural uranium as fuel. However, because the moderation properties of heavy water are not as good as light water, the volume ratio of moderator to fuel is five to eight times that of an LWR. Thus, the CANDU core is larger than a light-water reactor (LWR) core generating the same power. This results in a lower core power density for CANDU 3. In addition, the CANDU 3 core is neutronically loosely coupled, which results in xenon-induced flux tilt that requires a relatively complicated computer-operated spatial flux control system.

As in LWRs, CANDU 3 fuel elements consist of pressed and sintered uranium dioxide pellets enclosed in a zirconium cladding. Each CANDU 3 fuel bundle is about 20 inches long, consists of 37 fuel compacts, and is loaded into each of the 232 horizontal fuel channels. Each of the 232 horizontal fuel channels consists of a pressure tube concentrically placed inside a "calandria" tube. The pressure tubes form part of the reactor coolant system pressure boundary. Because of the low excess reactivity associated with a natural uranium core, the CANDU design must be fueled on a continuous basis during power operation by an automatic fueling machine. On-line fueling is the primary means of changing reactivity in the CANDU 3.

For the CANDU 3 design, heavy-water coolant flow through the core is unidirectional, thereby facilitating on-line fueling from one enu of the reactor with a single fueling machine. The primary system operating pressure (nominally 9995 kPa/1435 psig) is maintained by a pressurizer connected to one of the outlet headers. The CANDU 3 light-water secondary system is similar to that of a pressurized-water reactor (PWR).

The fuel channel assemblies are enclosed in a horizontal, cylindrical vessel called a calandria that contains the low-temperature (60 °C/140 °F), low-pressure, heavy-water moderator. The calandria vessel, in conjunction with the integral end shields, supports the horizontal fuel channel assemblies and the vertical and horizontal reactivity control unit components. The CANDU 3 utilizes four reactivity control systems for reactor control and shutdown during normal operation, and two redundant and diverse safety-grade shutdown systems are used for reactor shutdown following a transient. A separate moderator heat removal system ensures that the moderator remains subcooled.

All systems in the CANDU 3 design are assigned to one of two groups — either Group 1 or Group 2. The systems of each group are capable of shutting down the reactor, maintaining cooling of the fuel, and providing plant monitoring capability in the event that the other group of systems is unavailable. Group 1 systems are those primarily dedicated to normal power production. The Group 2 systems include four special safety systems and other safety-related systems. These maintain plant safety in the event of a loss, or partial loss, of Group 1 systems, and mitigate the effects of accidents, including the design-basis earthquake. The Group 1 and Group 2 systems are, to the greatest extent possible, located in separate areas of the plant. CANDU 3 employs two fast-acting, redundant, and diverse Group 2 shutdown systems, separate from the Group 1 reactor regulating system. Shutdown System No. 1 (SDS1) consists of 24 vertically inserted control rods. Shutdown System No. 2 (SDS2) consists of six horizontal nozzles through which a gadolinium nitrate solution is injected. Both shutdown systems inject into the low-pressure moderator, precluding a rod ejection accident. In addition to the two shutdown systems, the remaining special safety systems are a containment and an emergency core cooling system (ECCS).

The CANDU 3 containment system includes a reinforced-concrete containment structure with a reinforced-concrete dome and an internal steel liner. The containment is designed with a test acceptance leakage rate of 2 percent per day. The ECCS supplies light-water coolant to the reactor in the event of a loss-of-coolant accident. Each of the four safety systems is required to demonstrate during operation, a dormant unavailability of less than 10^{-3} or about 8 hours per year, and be physically and functionally separate from the normal process systems and from one another. The CANDU 3 shutdown cooling system is designed to remove heat from the heat transfer system at nominal operating temperature and pressure.

B. MHTGR (MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR)

Development History

The modular high-temperature gas-cooled reactor (MHTGR) was proposed to NRC by the U.S. Department of Energy (DOE) in 1986 in response to the Commission's Advanced Reactor Policy Statement (51 FR 24643). A preliminary safety information document (PSID) and 12 amendments were submitted between October 1986 and March 1992. The PSID and 10 of the amendments were reviewed by the Office of Nuclear Regulatory Research and a draft preapplication safety evaluation report (PSER) was issued by NRC in March 1989. The Energy Policy Act of 1992 requires DOE to submit a preliminary design approval application by September 30, 1996.

The first commercial gas-cooled reactors were the graphite-moderated, carbon dioxide-cooled Magnox reactors developed in the early 1950s in the United Kingdom and France. In the United States, gas reactor development resulted in the 40-MWe Peach Bottom Unit 1, which operated from 1967 to 1974, and the 330-MWe Fort St. Vrain plant, which operated from 1976 to 1989. There have been about 50 gas-cooled reactors in the world totaling about 1000 reactor-years of operation. In this total, there are about 50 reactor-years of experience with high-temperature gas-cooled reactors (HTGRs).

The BISO and TRISO (trade names) multilayered microsphere fuel form is used in HTGRs. The BISO fuel form, a fuel kernel with two major layers, was used in Peach Bottom Unit 1; and the TRISO fuel form, a fuel kernel with four major layers (including a silicon carbide layer), was used at Fort St. Vrain. The TRISO fuel form provides higher fuel integrity requirements than the BISO fuel and is the reference fuel for the MHTGR. DOE maintains agreements with Germany and France for the exchange of technical information concerning the integrity of the reference MHTGR fuel, and experiments will be conducted in France. As part of DOE's Technology Development Program for the MHTGR, post-irradiation testing of development fuel at Oak Ridge National Laboratory is being performed, and a technical information exchange agreement was established with Japan, which is building an experimental HTGR.

Major trends in recent HTGR designs, including the MHTGR, are the following: (1) increased system pressure, (2) steel pressure vessels for the smaller HTGRs, including the MHTGR, and prestressed concrete reactor vessels for larger HTGR designs, as Fort St. Vrain, (3) proposed greater fuel integrity, with a 6×10^{-5} fraction of failed fuel assumed for the MHTGR, and (4) lower enriched uranium fuel.

Design Description

The standard MHTGR plant comprises four reactor-steam-generator modules and two steam-turbine-generator sets. Each module is designed for a thermal output of 350 MWt. The standard plant configuration is designed to produce a total plant electrical output of 540 MWe. The low-power density (5.9 watts/cc) reactor core is helium cooled and graphite moderated, and uses ceramic-coated (four major layers) microspheres in an organic-bonded cylindrical compact as the fuel. The core design is intended to provide a large negative Doppler coefficient to shut down the reactor with heatup. The microsphere fuel design is stated to allow fuel temperatures as high as 1593 °C/2900 °F without significant fission product release. The compacts are placed in small vertical holes in the hexagonal graphite block fuel assemblies. The fuel assemblies are cooled through passages in the blocks. There are about 660 graphite blocks in the 66-column, annular core region between the inner and outer reflector regions. The helium is a single-phase coolant that is chemically and neutronically inert.

The MHTGR has a below-grade, safety-related reactor building, containing the reactor and steam generator vessels. The core is in a steel vessel located, with the steam generator, in the reactor building below ground to reduce seismic loads. The reactor vessel is above the steam generator vessel to prevent natural circulation and is connected to this vessel by a horizontal crossduct vessel. The reactor and steam generator vessels are in separate cavities. The secondary-side water is superheated in the steam generator. The core outlet helium temperature is about 704 °C/1300 °F and the steam outlet temperature is about 540 °C/1005 °F. The secondary-side pressure is higher (about 17,338 kPa/2500 psig) than that on the primary side (about 6479 kPa/925 psig), so water would leak into the helium coolant should a steam generator tube leak or fail.

Reactor protection is provided by two safety-related reactor protection systems (control rods and boron carbide balls), which are diverse and redundant, and one non-safety-related system (control rods). The non-safety-related system is independent from and redundant to the safety-grade systems. The equilibrium shutdown core temperature would be approximately 121 °C/250 °F, the design temperature for refueling.

The safety-related reactor cavity cooling system (RCCS) is a set of panels surrounding the reactor vessel with a header connection to four inlet and exhaust ports above ground. This allows hot air to rise, thus removing heat transferred from the reactor vessel while cold air is drawn from outside into the panels. The system (1) is entirely passive with no moving components, (2) is always operating, (3) automatically responds to rising temperatures through thermal radiation and natural circulation, and (4) has flow path redundancy to the cooling panels through a cross-connected header. In addition, there are two other non-safety-related, active heat removal systems: (1) the shutdown cooling system in the bottom of the reactor vessel and (2) the main circulator/steam generator in the primary cooling loop. The non-safety-related systems are not relied upon for accident safety analyses.

The multiple barriers to fission product release are the coated fuel microspheres, the graphite blocks, the ASME Code reactor coolant pressure boundary (RCPB), and the containment. The containment is the reactor building below ground and has containment isolation valves on the steam generator main steam and feedwater inlet piping. It will not retain the gases from a rapid RCPB depressurization, but is designed to have a leak rate of less than 100 percent per day after initial depressurization.

C. PIUS (PROCESS INHERENT ULTIMATE SAFETY) REACTOR

Development History

The process inherent ultimate safety (PIUS) reactor is being designed by ASEA Brown Boveri Atom (ABB-Atom). The concept evolved in the early 1980s from an extension of then ABB-Atom's low-temperature district heating design. In October 1989, ASEA Brown Boveri/Combustion Engineering (ABB-CE) requested a licensability review of the PIUS design in accordance with NUREG-1226, and in May 1990, ABB-CE submitted the PIUS preliminary safety information document (PSID) for staff review. ABB-CE plans to apply for design certification of the PIUS design sometime in 1994 or 1995, assuming a favorable preapplication review.

The PIUS design concept has already undergone tests related to the design principles. ABB-Atom has completed testing using the MAGNE Test Rig to simulate such PIUS parameters as diffusion and mixing across the primary loop/pool boundary with consideration for effects of turbulence, stratification, and migration of boron, among other effects. Large-scale tests of such PIUS design principles as flow and density lock operation were done at the ATLE Test Rig. These tests were used to validate the RIGEL code to calculate the design's safety and transient performance. ATLE was a full-height simulation of the PIUS pool. Other tests of the PIUS design principles have been carried out at MIT and TVA, and additional large-scale tests and a larger test rig are planned to be started this year for the purpose of design optimization, as well as special component testing. It is planned that this larger test rig will serve as the basic test facility for developing data for the detailed design and verification.

Design Description

PIUS is a 640-MWe advanced pressurized-water reactor (PWR) design with four loops. It relies on thermal-hydraulic effects to accomplish the control and safety functions that are usually performed by mechanical means. The safety-grade reactor heat removal system for the PIUS design is completely passive and is always in operation. The PIUS design consists of a vertical hollow cylinder, the reactor module, which contains the reactor core. The reactor module is submerged in a large concrete reactor vessel containing 3,300 m^3/870,000 gallons of highly borated water. The reactor module is open to the borated pool at the bottom and at the top of the reactor module. At these two openings, density locks keep the borated pool water from the reactor module during normal operation. Under normal operations, the primary loop reactor water flows up through the core, out of the top of the reactor module to the steam generators, and is pumped back into the bottom of the reactor module, bypassing both the top and bottom density locks. There is no physical flow barrier in the density locks between the primary loop and the borated pool; however, the difference in density between the primary loop reactor water and the cooler borated pool water provides a relatively stationary interface. When sufficiently upset during transient conditions, such as loss of flow or a power mismatch, the density difference is overcome and the borated water flows into the core and shuts down the reactor. A natural circulation flow path is then established from the borated pool through the lower density lock, up through the core, and back into the borated pool through the upper density lock for long-term shutdown cooling. Borated water can also be injected into the core through a redundant and diverse scram system consisting of lines which have been added between the reactor pool and the inlet to the reactor coolant pumps in each loop. These lines contain two "scram" valves, in parallel, that are normally closed. Upon receipt of a scram signal, the valves open and the imposed pressure difference induces highly borated water from the pool to flow to the pump inlet. The pumps continue to operate, thereby injecting highly borated water into the primary circulation loop. Unlike most reactors, PIUS does not employ mechanical control rods for regulating reactivity. Reactivity is controlled by the boron concentration and temperature of the primary loop reactor water.

Other aspects of the PIUS design are similar to the passive LWRs being considered by the staff (AP-600 and the SBWR). Although PIUS is a PWR, its operating pressure (8997.6 kPa/1,305 psi) is close to that of a BWR. The proposed containment for the PIUS design is integral with the reactor building, similar to the ABWR and SBWR. Leak rate has been defined as not to exceed 1 volume percent per day at a design pressure of 280.6 kPa/26 psig. The acceptance leakage value is expected to be 0.5 percent at design pressure.

D. PRISM (POWER REACTOR INNOVATIVE SMALL MODULE)

Development History

The United States Department of Energy (DOE) selected the power reactor innovative small module (PRISM) design as the advanced liquid-metal reactor (ALMR) design to sponsor for NRC design certification. The conceptual design for PRISM was developed by General Electric (GE) Company in conjunction with an industrial team of commercial engineering firms. Research and development support is being supplied by the Argonne National Laboratory, Energy Technology Engineering Center, Hanford Engineering Development Laboratory, and Oak Ridge National Laboratory. In addition, a steering group of utility representatives was involved in the PRISM design effort.

DOE chose to sponsor the PRISM design as part of its National Energy Strategy because of the design's potential for enhanced safety through the use of passive safety systems and greater safety margins, reduced cost through modular design and construction, and possible future development of an actinide-recycling capability. Although this last alternative has not yet been proposed in the current application, DOE has supported studies evaluating the use of actinides separated from spent fuel in an advanced liquid-metal reactor (ALMR) fast-flux core.

The PRISM design has considered liquid-metal reactor (LMR) experience to date developed both nationally and internationally in terms of systems and components design, reliability data, and safety assessments. This experience consists of operation of a number of facilities such as, EBR-II, Phenix, the Fast Flux Test Facility (FFTF), the Joyo reactor in Japan, and others.

The PRISM Preliminary Safety Information Document (PSID) was submitted to the NRC for review in November 1986, and the results of an early NRC staff review was the draft PSER (NUREG-1368) issued in September 1989. In order to obtain NRC approval of its prototype, DOE plans to apply for preliminary design approval in 1995. DOE also plans to apply for standard design certification in 2003 after a prototype demonstration. These plans are based on the current DOE goals to demonstrate the commercial potential for the ALMR by 2010, as called for in the Energy Policy Act of 1992.

Design Description

The PRISM plant design consists of three separate power blocks each made up of three reactor modules. Each module has a thermal output of 471 MWt and an electric output of 155 MWe for a total (plant) output of 1395 MWe. The PRISM design contains three turbines, each supplied from a power block. Options for one or two power blocks are possible. PRISM operates at much higher temperatures than current LWRs which will require a rigorous evaluation of the effects of creep and creep rupture on reactor vessel and systems. The PRISM design also relies on a highly automated and complex control system utilizing digital processing. The reactor module consists of the containment system, the reactor vessel, the core, and the reactor's internal components. The reactor vessel encloses and supports the core, the primary sodium coolant system, the intermediate coolant system heat exchangers (IHXs), and other internal components. The vessel is located just inside the containment vessel, which is located below grade in the reactor silo. The reactor vessel is penetrated only in the closure head. The head is supported by the floor structure, and the floor structure is supported by seismic isolator bearings to reduce horizontal movement during seismic events. The upper head of the reactor vessel is the closure head. The closure head also supports the intermediate heat exchangers (IHXs) and the electromagnetic (EM) pumps.

The main components of the nuclear steam supply system (NSSS) in PRISM are the reactor module, primary sodium loop, EM pumps, IHX, intermediate sodium loop and steam generators (SGs). The primary sodium loop is contained completely within the reactor vessel, which is hermetically sealed to prevent leakage of the primary coolant. The EM pumps provide the primary sodium circulation. Synchronous machines provide flow coastdown capability to the EM pumps. Flow coastdown is very important for preventing sodium boiling during a loss of EM pump power without reactor scram. Reactor-generated heat in the primary loop is transferred through the IHX to the intermediate heat transfer system (IHTS). IHTS sodium is circulated by a centrifugal pump. The IHTS operates at a higher pressure than the primary loop so that, in case of a tube rupture in the IHX, the sodium would not flow out of the reactor vessel. A pressure of approximately 204.7 kPa/15 psig is used to ensure a minimum 69.5 kPa/10-psi positive pressure differential across the IHX from the IHTS to the PHTS is maintained. A sodium-water reaction protection system mitigates the effects of reactions between IHTS sodium and water in the SG.

The reference fuel for the ALMR is a uranium-plutonium-zirconium (U-Pu-Zr) alloy. The ferritic alloy HT9 is used for cladding and channels to minimize swelling caused by high burnups. The PRISM core is a heterogeneous arrangement of driver fuel and blankets.

The PRISM core is designed so that the net power reactivity feedback is negative in all ranges of operation, in all transients, and in all accidents not involving voiding. For certain very-low-probability accident scenarios involving sodium boiling, a positive void coefficient dominates and a net positive feedback can occur. In all other situations without extensive voiding, an increase in temperatures produces negative feedbacks from Doppler and thermal expansion of the core and related structures that dominates the positive moderator density coefficient. The net negative temperature coefficient is so large that analyses predict all non-boiling transients and accidents to be terminated by the temperature feedback reactivity at temperatures low enough to not threaten fuel or vessel integrity. This passive shutdown function allows the reactor to sustain all non-boiling transient scenarios without damage, even with a failure to scram.

There are six control rods in the main reactivity control and shutdown system. Inserting any one of the six will shut the core down. The control rods can be inserted using (1) the plant control system (PCS) for normal insertion, (2) the safety-grade reactor protection system (RPS) for rapid insertion, and (3) gravity drop into the core. If both the normal and safety-grade systems fail, the operator can activate the ultimate shutdown system (USS) which sends boron balls into the central location of the core causing shutdown independently of the control rods. The PRISM design also includes passive mechanisms for controlling reactivity: three gas expansion modules (GEMs) consisting of tubes, closed at the top and open at the bottom, and filled with helium. If the pumps are running, the static pressure is high, causing the sodium level to rise to a high point in the GEM. However, with the pumps off, the static pressure and sodium level drop, which increases neutron leakage. The reactivity change provided by the GEMs between these two states is about -70¢.

Normal shutdown cooling is achieved with the non-safety-grade condenser. If the condenser becomes unavailable, the safety-grade reactor vessel auxiliary cooling system (RVACS) is used for RHR. The RVACS cools the containment vessel by means of natural air circulation. The design-basis RVACS event assumes that the normal and auxiliary heat removal systems, as well as the intermediate heat transport system (IHTS) sodium, are lost immediately following reactor and primary EM pump trips. The preapplicant's analysis shows that the RVACS heat removal rate is sufficient to maintain fuel temperatures within acceptable limits, and temperatures of the internal structures within the reactor vessel under American Society of Mechanical Engineers (ASME) Level C conditions. The PRISM design also contains the non-safety-grade auxiliary cooling system (ACS) to assist the RVACS. The ACS uses natural circulation within the steam generator (SG) to remove heat indirectly from the reactor vessel, and the natural circulation of air to cool the SG, with heat rejected directly to the atmosphere. The ACS can be used in combination with the RVACS to reduce the cooldown time. Some of the inherent safety characteristics of the PRISM design with respect to RHR are (1) the favorable combination of viscosity, thermal conductivity, and vapor pressure associated with the use of sodium to remove heat, (2) the ability to operate at essentially ambient pressure, thus reducing the pressure exerted on the coolant system boundaries, and (3) operation far below the sodium boiling temperature, thus obtaining the operational and analytical simplicity associated with a single-phase coolant.

COMMISSION PAPERS AND OTHER NRC DOCUMENTS PERTINENT TO TREATMENT OF ADVANCED REACTOR AND CANDU 3 POLICY ISSUES

- SECY-86-368, "NRC Activities Related to the Commission's Policy on the Regulation of Advanced Nuclear Power Plants," December 10, 1986
- SECY-89-350, "Canadian CANDU 3 Design Certification," November 21, 1989
- SECY-90-016, "Evolutionary LWR Certification Issues and their Relationship to Current Regulatory Requirements," January 12, 1990
    - Draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light Water Reactors and Their Relationship to Current Regulatory Requirements," February 27, 1992
    - Draft Commission paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light Water Reactor Designs," July 6, 1992
- SECY-90-055, "PIUS Design Review," February 20, 1990
- SECY-91-074, "Prototype Decisions for Advanced Reactor Design," March 19, 1991
- SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions," May 31, 1991
- SECY-91-202, "Departures From Current Regulatory Requirements in Conducting Advanced Reactor Reviews," July 2, 1991
- SECY-92-393, "Updated Plans and Schedules for the Preapplication Reviews of the Advanced Reactor (MHTGR, PRISM, and PIUS) and CANDU 3 Designs," November 23, 1992
- NUREG-1338, "Draft Preapplication SER for the MHTGR," March 1989
- NUREG-1368, "Draft Preapplication SER for PRISM," September 1989
- NUREG/CR-5261, "Safety Evaluation of MHTGR Licensing Basis Accident Scenarios," April 1989
- NUREG/CR-5364, "Summary of Advanced LMR Evaluations-PRISM and SAFR," October 1989
- NUREG/CR-5514, "Modeling and Performance of the MHTGR Reactor Cavity Cooling System," April 1990
- NUREG/CR-5647, "Fission Product Plateout in the MHTGR Primary System," April 1991

- NUREG/CR-5815, "Evaluation of 1990 PRISM Design Revisions," March 1992
- NUREG-1226, "Development and Utilization of the NRC Policy Statement on the Regulation of Advanced Nuclear Power Plants," June 1988
- NUREG-0444, "BEIRMOD, a Computer Program for Calculating the Effects of Exposure to Ionizing Radiation," May 1978