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Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: REVIEW OF THE DRAFT SAFETY EVALUATION REPORTS ON THE GE ADVANCED BOILING WATER REACTOR DESIGN

During the 383rd and 384th meetings of the Advisory Committee on Reactor Safeguards, March 5-7 and April 2-4, 1992, we discussed the Draft Safety Evaluation Reports (DSERs) on the Advanced Boiling Water Reactor (ABWR) design which is described by GE Nuclear Energy (GE) in its Standard Safety Analysis Report (SSAR), as amended, and for which GE has applied for design certification in accordance with 10 CFR Part 50, Appendix O. The DSERs which are the basis for this report were sent to the Commissioners for information as six SECY papers (SECY-91-153, 235, 294, 309, 320, and 355). These generally cover the SSAR and its first eighteen amendments. Our Subcommittee on Advanced Boiling Water Reactors discussed these papers with representatives of GE and the NRC staff during its meetings on September 18 and October 23, 1991 and January 23-24 and February 20-21, 1992. We also had the benefit of the documents referenced.

Our first report to you concerning the DSER for this project was dated November 24, 1989. That report conveyed our comments on Module 1 of the design (former GE designation). We also sent a report to you on July 18, 1991, outlining several ABWR design concerns that developed during subsequent review.

We note a marked improvement in the quality of the staff's DSER evaluations since our November 24, 1989 report. The staff reviewers appear to be following the guidance outlined in the applicable Standard Review Plans (SRPs) to the extent possible, and they are asking good in-depth questions in most areas.

The SECY-91-161 schedule indicates that the Final Design Approval (FDA) is to be issued before the end of Calendar Year 1992. If we are to provide our final report on this subject in December 1992, it will be necessary that we receive a complete and final SER no later than early September 1992. There are now more than three hundred open items in the DSERs, many of which are major. In addition, there is a number of important policy issues which are unresolved. With the staff programs in place, it is probable that these issues can be resolved. However, this is a large undertaking, and we have concerns about whether it can be accomplished on the schedule now indicated.

In the course of our review, we have identified technical issues for which resolutions should be achieved before we write our final report. These are listed and discussed as follows:

1. Control Building Flooding

The proposed ABWR plant design locates the Reactor Building Cooling Water (RBCW) System at the lowest elevation in the control building, with the essential 250 V dc battery rooms and the main control room at a higher elevation, but still below ground.

Our concern with this arrangement is the potential for control building flooding due to an unisolated break in the Reactor Service Water (RSW) System which provides cooling water from the Ultimate Heat Sink (UHS) to the RBCW System. The proposed UHS is a ground-level spray pond which we assume to be at building grade and likely to contain sufficient water to flood the control building.

The staff should obtain sufficient information on the interface and conceptual design of the RSW System and UHS to support an adequate evaluation of the flooding potential. The staff's evaluation should include consideration of isolation valve arrangements, the feasibility of and time available for response, and the assumption of a single active component failure during the response. The design information and flooding analysis should be included in the SSAR.

2. Adequacy of Physical Separation

Pipe breaks, internal plant flooding, and external events such as fire are of major concern if their effects cannot be confined in order to protect required safe-shutdown equipment. We believe that the key to confinement is the provision of appropriate separation barriers. However, a classical barrier such as the 3-hour-rated fire barrier wall and its penetrations (e.g., doors and dampers) may not, of itself, be sufficient to ensure separation under (a) the combined effects of pressure, heat, and smoke from a fire, and the flooding which results from fire mitigation, (b) the effects of pipe whip, jet impingement, or compartment pressurization due to pipe breaks, or (c) the influx of water and hydrostatic pressure buildup due to internal floods.

We believe that the SSAR should describe and the staff should evaluate the adequacy of proposed separation barriers for the full range of events and conditions for which separation must be ensured. We continue to recommend that systems required for safe shutdown not share a common Heating, Ventilating and Air Conditioning (HVAC) System during normal plant operation. The secondary containment HVAC System for the ABWR is such a shared system.

3. Protection of Environmentally Sensitive Equipment

The ABWR makes extensive use of environmentally sensitive equipment (including solid-state electronic components) for essential protection, control, and data transmission functions. Such components are known to be susceptible to adverse environmental changes, particularly temperature extremes. We are concerned that a number of these components may be located in plant areas where postulated events such as pipe breaks, fire, internal flooding, or loss of room cooling may create an adverse environment. Such environments need to be identified in the SSAR to ensure appropriate environmental qualification of the equipment.

4. Review of Chilled-Water Systems

The ABWR uses large chilled-water systems to provide essential environmental cooling, which in turn includes cooling of the solid-state electronic components. Because there was no SRP for chilled-water systems, the staff used other guidance such as SRP Section 9.2.2 (Reactor Auxiliary Cooling Water Systems) when the safety evaluation was performed. However, this guidance is not appropriate for the evaluation of refrigeration systems.

The NRC staff needs to evaluate the performance of chilled-water systems under varying accident heat loads and during loss-of-offsite-power events, and to consider their ability to restart and function after a prolonged station blackout. The DSER sections which should evaluate the performance of large chiller packages do not address these issues. We believe they should.

5. Use of Leak-Before-Break Methodology

It is our understanding that GE will not propose the use of leak-before-break methodology for the ABWR standard plant. Thus, the DSER should be revised to ensure that consideration is given to pipe break effects for all systems and locations. This may introduce additional structural protection and environmental qualification requirements in the SSAR.

6. Use of Integral Low-Pressure Turbine Rotors

In our July 18, 1991 report to you, we recommended that the staff review the issues involved with the use of integral low-pressure (LP) turbine rotors. It is our understanding that this new design for LP rotors will be used for the ABWR. (Rotors of this type are being used in rotor replacement programs at currently operating plants.) The practice of turbine manufacturers has been to bore the centerline of this type of rotor to remove impurity inclusions. We were concerned that the use of unbored rotors was being contemplated. The Electric Power Research Institute (EPRI) has recently added a requirement in its Advanced Light Water Reactor Utility Requirements Document (URD) that LP rotors be center-bored.

7. Cavity Floor Area Beneath Reactor Vessel

The cavity area beneath the reactor vessel is sized to meet the EPRI URD specification of 0.02 m2/Mwt. The ABWR design includes flooding of the cavity. Little consideration has been given to how this should be accomplished. There is little evidence that the planned cavity area will lead to quenching following flooding or that the ABWR flooding plans will not lead to ex-vessel steam explosions. Further attention needs to be given in the SSAR as to when and how fast the cavity should be flooded in order to avoid exacerbating a core-melt accident if it should occur.

8. Adequacy of the ABWR PRA

It is impossible to determine whether the PRA submitted by the applicant will be adequate for a safety determination absent information on how it is to be used by the staff. In our February 14, 1992 report to the Commission on the Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews, we commented on the need for guidance on the use of PRA in the review of new plant designs. At this point the applicant has submitted a PRA, a contractor has performed an extensive review, and the staff has prepared a DSER. However, the use of the PRA in the design certification process is still undefined.

Presumably, the results of the PRA will be used in the course of the staff's determination that the design is expected to produce a nuclear power plant that has an appropriate response to severe accidents. In the Severe Accident Policy Statement, the Commission indicated that a PRA would be required for each new design, and that the results of this PRA would be part of the information which would guide the staff in its determination that a design is adequate to deal with severe accidents. The policy statement published in the Federal Register of August 8, 1985, also states that "Accordingly, within 18 months of the publication of this Severe Accident Policy Statement, the staff will issue guidance on the form, purpose and role that PRAs are to play in severe accident analysis and decision making for both existing and future plant designs...." The Statement says further, "The PRA guidance will describe the appropriate combination of deterministic and probabilistic considerations as a basis for severe accident decisions."

The staff has yet to produce the promised guidance. We urge that the staff formulate a set of criteria that it plans to use in making severe accident decisions. This should include the way in which the results of a PRA are to be used in the process (not just whether the PRA has been done properly).

9. Containment Hydrodynamic Loads

Air-clearing loads on containment structures are the result of a complex process resulting from the drywell air being forced into the wetwell by the primary system blowdown. The water in the vent system is pushed down and out until the horizontal vents are cleared. The water-clearing process produces a jet of water into the suppression pool which causes a load on the outer part of the wetwell wall. This water clearing is followed by an air-steam mixture which creates a large bubble as it exits into the pool. The steam condenses but the air expands forcing the water above it up into the wetwell air space. The wetwell air space is compressed due to the momentum of the water in the layer above the bubble.

The wetwell air space will be subjected to an energetic two-phase eruption as a result of the air-clearing process. The vacuum breakers which are in the vicinity will be exposed to this environment unless protected. The SSAR should describe what the environment will be and what protective measures, if any, are needed to ensure survival of the vacuum breakers. If a vacuum breaker does not close, the suppression pool is bypassed and the wetwell/drywell pressures will rise at a rate dictated by the capability of some means other than the suppression process (e.g., containment sprays) to remove heat and condense steam. The SSAR should contain an analysis of such a situation.

The early work to address problems arising from analyses of the Mark I, II, and III containments is not sufficient to address similar processes that will occur following a LOCA in an ABWR containment. The ABWR is different for two reasons: (a) the volume of the wetwell air space in the ABWR is approximately that of a Mark II, and (b) the impact of the air-clearing loads will be alleviated somewhat because the expected blowdown flows are much smaller than those expected in a Mark I or Mark II. Nevertheless, the combination of a much smaller wetwell and the lower mass flow from the break have not received sufficient attention to be written off by the staff or GE without further analysis or experimental investigation. We are not aware of any testing of the ABWR type geometry. We believe there are sufficient differences in both geometry and LOCA characteristics to require further evaluation of the air- clearing phase of the LOCA by more extensive analysis and/or experimental investigation.

10. Adequacy of SSAR Treatment of the Reactor Water Cleanup System

We performed a review of the Reactor Water Cleanup (RWCU) System using our own staff. This system was chosen because it is a non-safety system located outside of primary containment, but inside the building which houses engineered safety features. It uses pipes up to 8-in. nominal diameter whose rupture would result in a LOCA and a source of serious environmental disruption in the building. This system is not seismically qualified or built to quality assurance standards.

Our review identified a number of deficiencies in the SSAR, some of which are listed below:

There is little useful information presented in the SSAR that describes how the Japanese codes and standards used for the RWCU System design can be converted to domestic design standards. The Quality Group classifications for certain portions of the RWCU System are inconsistent with the Japanese code-related classifications shown on the Piping and Instrumentation Diagrams. The Safety

Class/Quality Group transition between the piping inside primary containment and that outside primary containment is not in accordance with ANSI/ANS safety class standards for BWR fluid systems.

The questionable ability of system isolation valves to close under large-break-LOCA conditions has been the subject of extensive NRC testing and a Generic Letter (GL 89-10). However, the SSAR specifies no special performance requirements for these valves.

The safety-grade leak detection and isolation system which actuates the system isolation valves was not described in detail sufficient to support an assessment of its adequacy.

The ABWR PRA did not evaluate as initiating events RWCU System line breaks (or other LOCAs) outside the primary containment. The exclusion of these breaks was based erroneously on an analysis of the effects of suppression pool bypass events on overall risk. However, the analysis failed to take into account that the bypass path (e.g., RWCU System pipe break) could be the initiator for the core-damage event.

The PRA analysts took credit for the RWCU System as a heat removal system in all sequences where reactor pressure is assumed to remain high. The analysts assumed that the capacity of the non-regenerative heat exchanger (NRHX) is adequate to remove the decay heat. The capacity appears to be adequate; however, our calculations indicate that the outlet temperatures on the RWCU System side and cooling water side of the NRHX would exceed the design limits for the piping. Furthermore, a temperature sensor between the NRHX and the RWCU System pumps in the present design would automatically isolate the NRHX on high temperature, making it unavailable.

The items mentioned above are among a number of issues that were identified. It is important for the staff to ensure that the shortcomings of the RWCU System and PRA related portions of the SSAR are not indicative of problems in the remainder of that report.

11. Plant Design Life and Aging Management

We recommend that the SSAR clearly define the scope of the 60-year design life for the ABWR and describe a program plan for achieving it. This program should include those aging management measures which are necessary to maintain the plant within its design basis throughout its design life. This program should specify the original design and application criteria and, where required, the projected refurbishment or replacement requirements with appropriate rationale. To the extent applicable, the lessons learned from the NRC's Nuclear Plant Aging Research Program as well as other aging research projects should be incorporated into this program.

We note that the EPRI URD (Volume II, Chapter 1, Paragraph 3.3) includes a requirement for a plant design life of "60 years without necessity for an extended refurbishment outage," and discusses the requirements for its achievement in Paragraph 11.3.

12. Station Grounding and Surge Protection

Chapter 8 of the ABWR SSAR defines the scope of and specifies the requirements for the electrical power systems. The scope is limited to the onsite electrical power systems and to the interface requirements with the offsite electrical power systems.

Notably absent are lightning protection, station grounding systems, and surge protection measures which are necessary to protect plant personnel and equipment during normal and abnormal conditions. These measures are required to eliminate or reduce electrical shock hazards to personnel, and to protect systems and equipment against damage or misoperation as the result of lightning strikes, switching operations, electrical arcs, short circuits, static electricity, etc. These protective measures and their interface requirements should be included in the SSAR.

The ABWR makes extensive use of sensitive solid-state electronic components for essential protection, control, and data transmission functions. These components should be protected from extraneous electrical impulses that will damage them or cause improper performance. To the extent practical, these components should be isolated from potential adverse signals that may be transmitted over control or data links from remote locations, meteorological stations, switchyards, etc.

We note that the EPRI URD (Volume II, Chapter 11, Item 9, "Electrical Protective Systems") addresses requirements for these systems. We recommend that these grounding, surge protection, and isolation features be included in the SSAR.

13. Corrosion Control for Structures

The SSAR should include an interface requirement for a corrosion control program to identify the potential for the corrosion of structures and components and to determine the corrective measures to be taken. The program should commence prior to the completion of the detailed design of building substructures and underground installations. The program should consider the potential for corrosion from galvanic direct currents which may flow as the result of copper ground mats on site, including the electrical switching stations' ground mats. The potential for corrosion of containment building substructures and liners should be considered. The mitigation measures may include coatings, wrappings, cathodic protection, electrical bonding, elimination of galvanic currents, or other mitigation means.

We do not expect to receive a separate reply to the above items if they are covered appropriately in the final SER. We will keep you informed of any additional concerns as our review proceeds.

Sincerely,

David A. Ward
Chairman

References:

1. GE Nuclear Energy, Standard Safety Analysis Report, "Advanced Boiling Water Reactor," Chapters 1 through 20 (Amendments 1 through 18)
2. SECY-91-153, dated May 24, 1991, for the Commissioners from James M. Taylor, Executive Director for Operations, NRC, Subject: Draft Safety Evaluation Report (DSER) on the General Electric Company Advanced Boiling Water Reactor Design Covering Chapters 1, 2, 3, 4, 5, 6, and 17 of the Standard Safety Analysis Report (SSAR)
3. SECY-91-235, dated August 2, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapters 1, 3, 9, 10, 11, and 13 of the SSAR
4. SECY-91-294, dated September 18, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapter 7 of the SSAR
5. SECY-91-309, dated October 1, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapter 19 of the SSAR, "Response to Severe Accident Policy Statement"
6. SECY-91-320, dated October 15, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Advanced Boiling Water Reactor Design Covering Chapter 18 of the SSAR
7. SECY-91-355, dated October 31, 1991, for the Commissioners from James M. Taylor, EDO, NRC, Subject: DSER on the GE Boiling Water Reactor Design Covering Chapters 1, 2, 3, 5, 6, 8, 9, 10, 12, 13, 14, and 15 of the SSAR
8. Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document" (Volume II)/ALWR Evolutionary Plant, Revision 3, Issued November 1991