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UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

November 24, 1989

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

Dear Mr. Taylor:

SUBJECT: MODULE 1 OF THE DRAFT SAFETY EVALUATION REPORT FOR THE ADVANCED BOILING WATER REACTOR DESIGN

During the 355th meeting of the Advisory Committee on Reactor Safeguards, November 16-18, 1989, we met with representatives of the Office of Nuclear Reactor Regulation (NRR) and the General Electric Company (GE) to discuss Module 1 of the staff's Draft Safety Evaluation Report (DSER) for the Advanced Boiling Water Reactor (ABWR) design. This matter was also considered by our ABWR subcommittee during several meetings, the latest on October 31, 1989. We also had the benefit of the documents referenced.

The staff's DSER relates to the GE application for final design approval (FDA) and design certification of the ABWR design. The DSER is scheduled for completion in four modules. Module 1 is the subject of this letter and addresses Chapters 4, 5, 6, and 17 of the ABWR Standard Safety Analysis Report (SSAR) and corresponding chapters of the Standard Review Plan (SRP), NUREG-0800. Our review of these chapters of the SSAR has been completed through Amendment 7.

A number of the SSAR and DSER sections included in the Module 1 chapters are presently missing and will be issued as SSAR revisions and supplements to the DSER. Even within the included sections, there are a number of open, unresolved, and confirmatory issues and incomplete interface requirements or other information that will delay completion of our review until the revisions and supplements are issued. Comments on such missing or incomplete information will be included with our review of future modules.

Our comments should not be considered complete until we have prepared a report to the Commission concerning the final integrated DSER, which is presently scheduled for late 1990. For now, we are providing the following comments and recommendations concerning Module 1.

GENERAL

1. The staff's ABWR licensing review bases letter to GE (Reference 2) states, "The degree of design detail necessary for providing an essentially complete design is to be that detail that is suitable for obtaining specific equipment or construction bids and to demonstrate

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conformance to the design safety limits and criteria." We believe that the level of design detail in Module 1 falls short of this requirement. For example, we find that while GE has committed to follow applicable codes, standards, and regulatory guides, they have developed internal specifications for materials used in the fabrication of pressure boundary components that have not been submitted for NRC review. We also find that a number of design details (such as those relating to design temperature and pressure and pipe size) are indicated on drawings in the SSAR as "to be established by others" or similar statements. Unless such information is included in the SSAR or other documents that are reviewed by the staff, it is clear that the level of design detail is inadequate. We recommend that the staff revisit the issue of what constitutes an "essentially complete" design. The staff should also consider the question of form and depth of reporting differences between the ABWR being designed for construction in Japan and the ABWR design being proposed for certification.

2. The SSAR chapters contain a number of sections for which there are no corresponding sections in the DSER or SRP, or the subjects of the DSER or SRP sections are different. Also, there are cases wherein the SRP contains sections that do not appear in the SSAR or DSER. We recommend that the DSER sections be referenced by number and title to the corresponding SSAR sections they evaluate. Differences, including the absence of any corresponding SRP sections, should be identified in the DSER.

CHAPTER 4 - REACTOR

3. The fine motion control rod drive system (FMCRDS) materials list discussed in SSAR Section 4.5.1.1 shows Stellite guide rollers and roller pins. Section 5.2.3.2.2.2 states that cobalt base alloys used for pins and rollers in the FMCRDS have been replaced with noncobalt alloys. The list of materials should be corrected.

4. We were told by GE that the design of the integral rod ejection support system for the FMCRDS has been changed from that described in SSAR Section 4.6.1. The staff should determine that their evaluation in the DSER is based on the revised design and the SSAR should be corrected.

CHAPTER 5 - REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5. The SSAR states that the automatic depressurization system (ADS) utilizes safety relief valves (SRVs) each of which is equipped with an air accumulator and check valve arrangement designed to ensure two actuations following failure of the air supply. Although not stated in the SSAR, GE indicated that the accumulators are backed up by the nitrogen supply system. This backup arrangement needs to be described in the SSAR together with how check valve operability will be ensured.

6. The specifications given in the SSAR for the materials of the primary pressure boundary do not meet current "good practice," or the practice GE says they would require in the construction of an ABWR--they should. To clarify this issue, the SSAR should contain answers to the following questions: (1) will the steel in the core beltline be forged rings or welded plate?; (2) will upper limits on sulfur content of the rolled plate in the pressure vessel be those given in the ASME Code SA-533, Specifications for Pressure Vessel Materials (0.04%) or lower values consistent with good modern practice (under 0.015% with shape control)?--an adequate level is specified for forged segments (ASME Code SA-508, Class 3, Specification for Quenched and Tempered Vacuum-Treated Forgings) and is available as an option in SA-533 but not called out by GE; and (3) what will be the upper limit on delta ferrite for cast stainless steel components? The Code's allowed value of 25% should be halved to substantially remove concern about long-term aging.

7. SSAR Section 5.3.3 states that design for vessel annealing is not required because the predicted value of adjusted RT_{NDT} does not exceed 200^{o} F. The DSER states that the integrity of the reactor vessel is ensured because the vessel may be annealed, if necessary. GE stated during our meeting that the vessel is not designed to be annealed. The DSER statement should be resolved with GE.

8. We believe that potential safety hazards (e.g., excessive internal pressure) associated with an uncleared electrical fault inside a reactor internal pump (RIP) should be analyzed and documented in the SSAR.

9. We were told by GE that motor restraint rods are provided to prevent ejection of an RIP. We believe that this important feature should be described in the SSAR and evaluated by the staff.

10. SSAR Section 5.4.6 states that the design basis for the Reactor Core Isolation Cooling (RCIC) system is only 30-minutes of operation during a loss-of-ac power event. We believe that a more complete discussion of the station blackout capability should be included in the SSAR. The DSER should include an evaluation of the 30-minute capability as an acceptable design basis.

11. The DSER contains no specific references to SSAR Sections 5.4.4-5, 5.4.9, and 5.4.12-14. These sections discuss feedwater piping, main steam line flow restrictors, isolation systems and piping, component supports, and valves. There are no comparably numbered sections in the SRP. It is not clear where the staff intends to report its evaluation of these important topics.

CHAPTER 6 - ENGINEERED SAFETY FEATURES

12. The design basis for the ECCS and the conclusions given about its performance do not include the ejection of an RIP (450 cm^2 break).

The rationale for excluding such an event as a design basis break should be discussed in the SSAR.

13. DSER Section 6.2.6 indicates that inflatable seals will be used for primary containment equipment and personnel air lock penetrations. We believe that an appropriate description of the seals and the air supply arrangement and reliability should appear in the SSAR. The discussion should include the capability of the seals to function under elevated pressure and temperature conditions for prolonged periods of time following a design basis accident.

14. There is a new section 6.5.5 (Pressure Suppression Pools as Fission Product Clean-Up Systems) in the SRP which does not appear in the SSAR or DSER. Why is this SRP section not being used for the ABWR?

CHAPTER 17 - QUALITY ASSURANCE

15. Chapter 17 of the SSAR is intended to describe how GE and its major technical associates (not mentioned by name in the SSAR but we assume to be Toshiba Corporation and Hitachi Limited) engage in the joint development and engineering of the ABWR design. The quality assurance programs used by the technical associates are not described or referenced in the SSAR. We believe they should be.

In conclusion, we believe that significant progress has been made by the staff in its review of the SSAR for the Advanced Boiling Water Reactor. A considerable amount of work remains to be completed before the FDA is issued as expected by the end of 1990. We will continue to review this work as the documentation becomes available.

Sincerely,

F J Remick

Forrest J. Remick
Chairman

References:
1. Letter dated August 17, 1989 from Charles L. Miller, Office of Nuclear Reactor Regulation, NRC, to Mr. Patrick W. Marriott, General Electric Company, enclosing Draft Safety Evaluation Report Related to the Final Design Approval and Design Certification of the Advanced Boiling Water Reactor, dated August 1989
2. Letter dated August 7, 1987 from Thomas E. Murley, Office of Nuclear Reactor Regulation, NRC, to Ricardo Artigas, General Electric Company, enclosing GE Advanced Boiling Water Reactor, Licensing Review Bases, dated August 1987
3. GE Nuclear Energy, Standard Safety Analysis Report, Advanced Boiling Water Reactor, Chapters 4, 5, 6, and 17