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The Honorable Ivan Selin
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Selin:

SUBJECT: REPORT ON SAFETY ASPECTS OF THE GENERAL ELECTRIC NUCLEAR ENERGY APPLICATION FOR CERTIFICATION OF THE ADVANCED BOILING WATER REACTOR DESIGN

During the 408th meeting of the Advisory Committee on Reactor Safeguards, April 7-8, 1994, we completed our review of the General Electric Nuclear Energy (GENE) application for certification of its U.S. version of the Advanced Boiling Water Reactor (ABWR) standard design. This final report is intended to fulfill the requirement of 10 CFR 52.53 that the ACRS "... report on those portions of the application which concern safety." During our review we had the benefit of discussions with representatives of GENE and the NRC staff. We also had the benefit of the documents referenced.

ABWR Application

The U.S. version of the ABWR standard design utilizes a significant portion of the detailed design information developed jointly by GENE, Hitachi, and Toshiba for the international version which is being built in Japan. The application for certification of the U.S. version was filed by GENE in September 1987 under the provisions of Appendix O to 10 CFR Part 50 and the NRC Policy Statement on Nuclear Power Plant Standardization (Ref. 1). The application was docketed in February 1988. In December 1991, GENE requested that the application be considered under 10 CFR 52.45. This request was made effective in March 1992.

The application is based on the ABWR Standard Safety Analysis Report (SSAR), which was submitted in modular form between September 1987 and March 1989. Since then it has been amended frequently, the last submittal for our review was Amendment 34 in March 1994. The application also includes the ABWR Certified Design Material (CDM). The CDM contains the design information from the SSAR that will become a part of the design certification rule. The CDM has been revised, the last submittal that we received was Rev. 2 in December 1993.

ABWR Design Description

The ABWR is a forced circulation boiling water reactor with a rated power of 3926 MWt. The reactor core consists of 872 8x8 fuel assemblies and 205 control rods. The reactor utilizes internal recirculation pumps and fine-motion control rod drives. It is located inside a steel-lined reinforced concrete pressure suppression containment which is enclosed by a reinforced concrete secondary containment, both of which are located in the Reactor Building. The Reactor Building also houses a standby gas treatment system, refueling area, main steam pipe tunnel, and essential systems for emergency core cooling, AC power (including diesel generators), and environmental conditioning.

The Control Building is located between the Reactor Building and the Turbine Building. The Control Building houses a continuation of the main steam pipe tunnel, the main control room, a computer facility, and essential systems for DC power, environmental conditioning, and cooling water. During emergencies, technical support is provided by the Technical Support and Operational Support Centers, which are located in the Service Building, which is immediately adjacent to the Control Building.

The Turbine Building houses equipment for power generation. Steam is supplied to an 1800 rpm turbine-generator which is oriented to minimize damage to safety-related equipment should a turbine failure occur. The Turbine Building also houses systems and equipment that provide various nonessential services for the plant. These include the standby combustion-gas-turbine generator, house boiler, air compressors, and systems for AC and DC power and environmental conditioning.

The Radwaste Building houses equipment for the collection and processing of radioactive waste generated by the plant. An underground pipe tunnel connects the Turbine and Reactor Buildings to the Radwaste Building.

The ABWR design includes a number of features that we believe will enhance safety relative to past BWR designs. Some of these features resulted from the use of PRA methodology by GENE in evaluating the ABWR design as it progressed.

The use of reactor internal pumps removes the large reactor recirculation piping and connections to the reactor vessel, thereby reducing the size of the largest loss-of-coolant accident (LOCA).

The use of a fine-motion control rod drive arrangement removes the scram discharge volume and associated piping, provides two reliable means for inserting the rods, and is intended to eliminate the rod drop and rod ejection accidents.

The Emergency Core Cooling System and supporting auxiliaries are arranged into three physically separated electrical and mechanical divisions, only one of which is needed for handling transients and virtually all accidents.

A combustion-gas-turbine generator is provided for enhanced on-site AC power capability.

An AC-independent reactor water addition feature, a depressurization system, lower drywell flooder, cavity floor spreading area, sacrificial layer of basaltic concrete, and containment overpressure protection system are provided to mitigate severe accidents.

The greatly increased application of digital control systems offers the potential for improved operator interface with the plant and the reliability of control and protection systems. In addition, the use of digital multiplexers and fiber optics reduces the amount of cabling in the plant thereby reducing the fire hazard.

The reactor vessel is fabricated using ring forgings that eliminate the need for beltline longitudinal welds. This, in combination with improved material specifications, reduces concern for reactor vessel integrity.

Chronology of ACRS Review

Our review of the ABWR application commenced after it was docketed in February 1988. The NRC staff issued a Draft Safety Evaluation Report (DSER) on the first module of the SSAR in August 1989 (Ref. 2). We reviewed this draft and reported our findings in November (Ref. 3). At that time we questioned, in particular, the adequacy of the level of design detail available for review and recommended that the staff revisit the issue of what constitutes an "essentially complete" design.

Subsequent to November 1989, our review activities focused on several ABWR-related design concerns including Control Building flooding, physical separation, environmental protection of sensitive equipment, performance of essential chilled water systems, use of leak-before-break methodology, use of integral low-pressure turbine rotors, and the capability of the floor area beneath the reactor vessel to cope with severe accidents. These preliminary concerns were brought to the attention of the NRC staff in our July 1991 report (Ref. 4).

During 1991 the DSER was completed by the NRC staff in the form of six SECY papers (SECY-91-153, 235, 294, 309, 320, and 355). These papers generally covered most sections of the SSAR through the first eighteen amendments, but contained numerous open items. We reported our findings in April 1992 (Ref. 5). In this report, we reconfirmed the preliminary concerns expressed in our July 1991 report and added several more including adequacy of the PRA, containment hydrodynamic loads, Reactor Water Cleanup System safety implications, plant design life and aging management, station grounding and surge protection, and corrosion control for structures.

In October 1992, the NRC staff issued a Draft Final Safety Evaluation Report (DFSER) (Ref. 6) covering the entire SSAR through Amendment 20. This draft superseded the six SECY papers. The final version of the staff safety evaluation report which we reviewed was the "Advance Copy of Safety Evaluation Report related to the certification of the Advanced Boiling-Water Reactor Design," dated December 1993 (Ref. 7). This copy covered the NRC staff review of SSAR information through about Amendment 32. Additional changes, including those which reflect Amendments 33 and 34, were reviewed by us as page changes to Reference 7.

Between February 1988 when the ABWR application was docketed and

April 1992 when we issued our report on the DSER, our ABWR subcommittee held numerous meetings to review the SSAR and the NRC staff safety evaluations. During this same period, our subcommittee on Improved Light Water Reactors held several meetings to review the Electric Power Research Institute (EPRI) Utility Requirements Document (URD) and associated NRC staff safety evaluations for the Advanced Light-Water Reactor (ALWR) evolutionary plant. (The EPRI URD prescribes ALWR design requirements from the utility industry perspective.) Meetings were also held by our subcommittees on Auxiliary and Secondary Systems, Computers in Nuclear Power Plant Operations, Human Factors, and Severe Accidents. These subcommittees reviewed a number of specialized aspects of the proposed ABWR design including those related to fire, digital control and protection systems, human factors, and severe accidents.

Between April 1992 and today, our ABWR subcommittee held additional meetings to review design features proposed beyond Amendment 20 of the SSAR and to review the DFSER and Reference 7. This review covered significant design changes in the SSAR (through Amendment 34) and closure of all open items in the DFSER. It also included a review of written responses by GENE to numerous questions and concerns raised by the subcommittee.

During this time our subcommittee on Improved Light Water Reactors held several meetings to complete its review of the EPRI URD. In addition, ABWR-related meetings were held by our subcommittees on Auxiliary and Secondary Systems, Computers in Nuclear Power Plant Operations, Human Factors, Severe Accidents, Safeguards and Security, and our Ad Hoc Subcommittee on Design Acceptance Criteria (DAC). We did not review most of the CDM portion of the application because we were assured by the NRC staff that it did not contain design features and requirements beyond those found in the SSAR. We did, however, review and comment (Ref. 8) on the viability of the DAC process as a suitable method for establishing future design acceptance requirements in certain areas (i.e., human factors engineering, radiation protection, piping design, and instrumentation and control). We also reviewed the CDM related to these DAC areas.

During our review of the ABWR SSAR, we considered the design-specific requirements which relate to the various evolutionary and advanced light water reactor policy, technical, and licensing issues included in SECY-90-016 (Ref. 9) and its successor, SECY-93-087 (Ref. 10). These issues incorporate staff positions that deviate from or are not embodied in current regulations. Their resolutions will become "applicable regulations" through incorporation into the design certification rule for the ABWR. We have commented previously (Refs. 11 and 12) concerning these issues.

ACRS Conclusion Concerning ABWR Safety

Based on the results of our review of those portions of the GENE ABWR application which concern safety, we believe that acceptable bases and requirements have been established in the application to assure that the U.S. version of the ABWR standard design can be used to engineer and construct plants that with reasonable assurance can be operated without undue risk to the health and safety of the public.

Additional comments by ACRS Members Carlyle Michelson and Charles J. Wylie are presented below.

Sincerely,

J. Ernest Wilkins, Jr.
Chairman

Additional Comments by ACRS Members Carlyle Michelson and Charles J. Wylie

Although the Committee has arrived at a favorable conclusion concerning ABWR safety with which we agree, it is our view that this report should discuss the resolution of various issues that were considered by the Committee (Refs. 4 and 5) prior to reaching the favorable conclusion. Some of the resolutions were based on findings that were unanticipated and led to significant design changes. We believe that these findings should be made available to those who must make the final safety and design certification decisions.

As an example, it was found that the rupture of an 8-inch pipe in the non-safety-grade Reactor Water Cleanup (CUW) System which is housed inside of secondary containment creates serious environmental disruption throughout the three separate divisional areas of secondary containment which house redundant portions of the Emergency Core Cooling System (ECCS). Since this 8-inch pipe contains reactor coolant at operating temperature and pressure, the break results in an immediate loss of reactor coolant until isolated and it requires an ECCS response. Steam from the break permeates the entire secondary containment because the divisional barrier doors are forced open by a buildup of steam pressure. This occurs before the primary containment isolation valves for the CUW system have time to close. A similar situation exists for the Reactor Core Isolation Cooling (RCIC) System; however, the resulting environmental conditions for most locations are bounded by those produced by the CUW 8-inch pipe break.

Since these pipe break events cannot be confined, GENE now proposes that safety-related equipment inside of the ABWR secondary containment be environmentally qualified for steam at 15 psig. and about 248°F. It is our view that this is an acceptable, although undesirable, alternative to a design which provides separation barriers and pressure relieving pathways that are capable of isolating a sufficient amount of ECCS equipment from the harsh environment. In addition, GENE has added a third break isolation valve in the 8-inch CUW supply line and located it inside of primary containment. This valve can be closed after the blowdown is over to ensure the interruption of any prolonged loss of ECCS water to secondary containment. It is needed only if both primary containment isolation valves fail to fully close due to the severe blowdown loads or other challenges common to both valves. The added environmental qualification and the third valve are new features.

References:

1. U.S. Nuclear Regulatory Commission, Policy Statement, 10 CFR Part 50, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987
2. Letter dated August 17, 1989, from Charles L. Miller, NRC Office of Nuclear Reactor Regulation, to 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, August 1989
3. ACRS report dated November 24, 1989, from Forrest J. Remick, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Module I of the Draft Safety Evaluation Report for the Advanced Boiling Water Reactor Design
4. ACRS report dated July 18, 1991, from David A Ward, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Concerns Related to the General Electric Advanced Boiling Water Reactor Design
5. ACRS report dated April 13, 1992, from David A. Ward, ACRS Chairman, to James M. Taylor, NRC Executive Director for Operations, Subject: Review of the Draft Safety Evaluation Reports on the GE Advanced Boiling Water Reactor Design
6. U. S. Nuclear Regulatory Commission, NUREG-1469, "Draft Final Safety Evaluation Report Related to the Design Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor," October 1992
7. U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, "Advance Copy of Safety Evaluation Report related to the certification of the Advanced Boiling-Water Reactor Design," December 1993
8. ACRS report dated January 14, 1994, from J. Ernest Wilkins, Jr., ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: Final Report on the Use of the Design Acceptance Criteria Process in the Certification of the General Electric Nuclear Energy Advanced Boiling Water Reactor Design Approval
9. SECY-90-016, dated January 12, 1990, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements
10. SECY-93-087, dated April 2, 1993, from James M. Taylor, NRC Executive Director for Operations, for the Commissioners, Subject: Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs
11. ACRS report dated April 26, 1990, from Carlyle Michelson, ACRS Chairman, to Kenneth M. Carr, NRC Chairman, Subject: Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements.
12. ACRS report dated April 26, 1993, from Paul Shewmon, ACRS Chairman, to Ivan Selin, NRC Chairman, Subject: SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to

Evolutionary and Advanced Light-Water Reactor (ALWR) Designs"