POLICY ISSUE
(Information)

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SECY-89-153

For: The Commissioners

From: Victor Stello, Jr.
Executive Director for Operations

Subject: SEVERE ACCIDENT DESIGN FEATURES OF THE ADVANCED BOILING WATER REACTOR (ABWR)

Purpose: In the statement of considerations of 10 CFR Part 52, the Commission asked the staff to advise them on the need for criteria that are different from or supplementary to current standards. The purpose of this paper is to inform the Commission of certain features of General Electric's (GE's) ABWR design that the staff believes will enhance safety and will satisfactorily address severe accident issues when the staff's review is complete.

Background: SECY-89-013 informed the Commission of recent staff decisions regarding design enhancements for three standardized evolutionary advanced light-water reactors (ALWRs) -- General Electric's ABWR, Westinghouse's RESAR SP/90, and Combustion Engineering's CESSAR System 80+ -- and the Electric Power Research Institute's (EPRI's) Utility Requirements Document.

Consistent with the Commission policy on severe accidents, General Electric has proposed additional design features to enhance the capability of the ABWR to prevent and mitigate severe accidents. These enhancements were developed by GE in the context of their commitments in the Advanced Boiling Water Reactor Licensing Review Bases (ABWR-LRB), dated August 7, 1987, to further reduce the core damage frequency as well as the potential for a large release for this design.

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Discussion: The severe accident issues to be dealt with during the reviews of the ABWR design are briefly discussed below. A certain number of these issues were previously discussed in SECY-89-013. The severe accident concerns as they relate to the other designs, Westinghouse's RESAR SP/90, and Combustion Engineering's System 80+, will be discussed with the Commission at a later date.

In the ABWR design, GE has committed to the following Commission guidance on severe accidents for future plants as codified by 10 CFR Part 52:

- Demonstration of compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f);
- Proposed technical resolution of the applicable Unresolved Safety Issues and medium- and high-priority Generic Issues; and
- A design-specific probabilistic risk assessment.

This paper is structured in such a manner to identify (1) certain major initiating events that could lead to a severe reactor accident, (2) ABWR design enhancements that reduce the probability of their occurrence, (3) important ABWR systems that would be used to mitigate a severe accident should one occur, and (4) other topics related to severe accidents and the ABWR design. The initiating events and severe accident issues identified have resulted from a thorough staff review of past severe accident studies and research applicable to the ABWR.

STATION BLACKOUT

The station blackout rule (10 CFR 50.63) allows utilities several design alternatives to ensure that an operating plant can safely shut down in the event that all ac power (offsite and onsite) is lost.

The staff believes that the preferred method of demonstrating compliance with 10 CFR 50.63 is through the installation of an installed spare (100 percent capacity) alternate ac power source (and auxiliaries) of diverse design (from a different manufacturer) that is consistent with the guidance in Regulatory Guide 1.155.

The ABWR design clearly goes beyond the requirements identified in the station blackout rule. The ABWR design includes three independent electrical divisions, each with high-pressure and low-pressure capability, each powered by a 100 percent capacity diesel generator, and each division capable of independently shutting the reactor down. Additionally, the ABWR design includes an alternate ac combustion turbine to back up the diesel generators. The design has a capability to survive a 10-hour blackout period utilizing the reactor core isolation cooling (RCIC) turbine and station batteries. Extended blackout capabilities are also provided by the ac-independent water addition system. This system allows for makeup to the reactor vessel following RCS depressurization by connecting a direct drive diesel fire pump or by connecting an external pumping source, such as a fire truck, to a yard standpipe.

INTERSYSTEM LOCA

Future ALWR designs can reduce the possibility of a loss-of-coolant accident (LOCA) outside containment by designing (to the extent practicable) all systems and subsystems connected to the reactor coolant system (RCS) to an ultimate rupture strength at least equal to the full RCS pressure.

The ABWR has been designed to minimize the possibility of an interfacing system LOCA in the following ways. The low pressure systems directly interfacing with the RCS are designed with 500 psig piping which provides for a rupture pressure of approximately 1000 psig. In addition, the high/low-pressure motor-operated isolation valves have safety-grade, redundant pressure interlocks. Also, the motor-operated emergency core cooling system (ECCS) valves will only be tested when the reactor is at low pressure. All inboard check valves on the ECCS will be testable and have position indication. Additionally, design criteria used by GE require that all pipe designed to 1/3 or greater of reactor pressure requires two malfunctions to occur before the pipe would be subjected to reactor system pressure. The pipe designed to less than 1/3 reactor pressure requires at least three malfunctions before the pipe would be subjected to reactor system pressure.

The staff believes that these features should provide sufficient margin for all high/low-pressure interfaces to eliminate the concern about LOCAs outside of containment at the high/low-pressure interface of systems connected to the reactor coolant system (RCS).

ANTICIPATED TRANSIENT WITHOUT SCRAM

The anticipated transient without scram (ATWS) rule (10 CFR 50.62) was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred. In this regard, the ABWR design has a number of features that reduce the risk from an ATWS event. The staff believes that the modest enhancements proposed by GE can further reduce the risk from an ATWS event. These features include a diverse scram system with both hydraulic and electric run-in capability on the control rods, a manual-operated standby liquid control system, and a recirculation pump trip capability. In addition, the scram discharge volume has been removed from the ABWR, eliminating some of the potential ATWS problems associated with the older BWR designs.

While the ATWS rule requires an automatically initiated standby liquid control system (SLCS), GE has concluded that the enhanced reliability of the reactor protection system negates the need for an automatic SLCS. General Electric has agreed to provide a reliability analysis of the SLCS to support this position. The staff believes the analysis should include an evaluation of the control room displays, emergency procedures, and the time available for operator action.

HYDROGEN CONTROL

The Commission's Severe Accident and Standardization Policy Statements stated that future designs should address the provisions of 10 CFR 50.34(f). The Commission's stated policy has been codified in 10 CFR Part 52 to require the technically relevant provisions be met. Specifically, in order that containment integrity be maintained, 10 CFR 50.34(f)(2)(ix) requires future designs to provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal water reaction. In addition, this system needs to be capable of precluding uniform concentrations of hydrogen from exceeding 10 percent (by volume), or an inerted atmosphere within the containment must be provided.

The ABWR design meets the requirements of 10 CFR 50.34(f)(2)(ix) by utilizing a nitrogen-inerted atmosphere within containment. Also, a hydrogen recombiner for design-basis accidents will be provided in the ABWR design.

VENTING

In order to preclude an irreversible rupture of the containment, the ABWR design will include a "hardened" wetwell vent capable of venting at pressures up to approximately 80 psig. Venting operations will require the use of dc power and pneumatic pressure to open the isolation valves. In venting situations, all fission products would be filtered through the suppression pool. The containment vent system will prevent containment failure due to overpressure and temperature.

CORE-CONCRETE INTERACTION - ABILITY TO COOL CORE DEBRIS

In the unlikely event of a severe accident in which the core has melted through the reactor vessel, it is possible that containment integrity could be breached if the molten core is not sufficiently cooled. In addition, interactions between the core debris and concrete can generate large quantities of additional hydrogen and other non-condensible gases.

The ABWR design has a number of features that would mitigate the effects of a molten core. The suppression pool surrounds the lower drywell cavity and would thereby prevent core debris from reaching the containment boundary and breaching its integrity. Also, the ABWR is designed with a lower drywell flooder and a cavity space sufficient to be able to disperse core debris at an energy level of 1 MWt/m^2. The flooder consists of a number of temperature-sensitive fusible plugs that allow suppression pool water to enter the drywell cavity when high temperature resulting from core debris occurs in the lower drywell. The horizontal vents to the suppression pool will remain covered in the event of lower drywell flooding. GE anticipates that any core-concrete interaction will be stopped when the suppression pool water quenches the molten core debris. By providing sufficient area to allow the core debris to spread to a shallow bed and by flooding the core debris, it is expected that the potential for extensive core-concrete interactions will be significantly reduced. In addition, even if limited core-concrete interactions continue, the overlying pool of water will mitigate the consequences of these interactions by scrubbing the fission products and cooling the gases released from the concrete.

SOURCE TERMS

Regulatory Guides 1.3 and 1.5 provide the staff's principal bases for implementing the requirements in 10 CFR Part 100 for the ABWR. The radiological "source term" in these guides is based, in part, on the 1962 "TID-14844 source term." From the outset, these in-containment source terms were widely acknowledged to be very conservative but were justified on the basis of the uncertainties associated with accident sequences and equipment performance at the time of promulgation (circa 1962).

The staff intends to continue using existing methodology for the licensing basis source term for ALWRs in order to confirm compliance with 10 CFR Part 100 with respect to design basis accidents. The ABWR design will provide dual remote control room ventilation intakes and will maintain a combined main steam isolation valve (MSIV) leakage rate of less than 100-150 standard cubic feet per hour (scfh). The staff will consider giving credit for the steam line condenser as a hold-up pathway, and to the acceptance of a single charcoal bed filter in the standby gas treatment system. This credit, however, is predicated on additional analysis by GE regarding these systems and subsequent staff review.

The staff is presently working with GE in order to agree on a design specific source term and a definition of containment failure that can be used to address the severe accident goals identified by GE. These goals limit the probability of occurrence of offsite doses in excess of 25 rem beyond a one-half mile radius from the reactor to less than 10^{-6}/year, and a containment conditional failure probability less than 10^{-1} weighted over credible core damage sequences.

PROBABILISTIC RISK ASSESSMENT (PRA)

10 CFR Part 52 requires a design-specific PRA based upon the bounding site parameters for the design. To ensure the design certification PRA assumptions are retained as part of the design and operation of the plant, the ABWR licensing bases will be incorporated into the design certification.

GE has committed to provide a level-3 PRA including full power, low power and refueling conditions and addressing internal events and a bounding external events analysis. GE has also agreed to provide in the ABWR standard safety analysis report (SSAR) the reliability and maintenance criteria that a future applicant must satisfy to ensure that the safety of the as-built facility will continue to be accurately described by the certified design. The ABWR SSAR is to include the key assumptions of the PRA and other PRA licensing commitments.

BWR THERMAL-HYDRAULIC STABILITY

The staff believes the issue of BWR stability, while not directly a severe accident concern for the ABWR, is an important topic relative to design certification and is therefore discussed in this paper.

In boiling-water reactors, thermal-hydraulic instabilities can cause oscillations that can result in violation of the minimum critical power ratio (MCPR) safety limits. In order to cope with the stability problem, GE has provided additional preventive and mitigative measures to the ABWR design. These include, in part, an automatic logic that prevents plant operation in the region of least stability, and selected control rod run-in that is automatically initiated to avoid stability concerns when a trip of two or more reactor internal pumps (RIPs) occurs. The ABWR design reduces the potential for the type of event that occurred at LaSalle.

Conclusion: The staff believes that the issues and associated ABWR design enhancements discussed in this paper reflect the experience that has been gained from the current generation of operating plants, and that they are in keeping with the Commission's policy that future designs for nuclear plants should reduce the risk from severe accidents. The staff believes that its review will confirm the effectiveness of these features in addressing the Commission's severe accident goals defined in the Severe Accident, Safety Goal, and Standardization Policy Statements as well as complying with 10 CFR Part 52 as it relates to design certification.

It is the staff's intention that equipment required to mitigate a severe accident but not required to mitigate a design-basis accident (DBA) (e.g., core debris quenching equipment) must be able to maintain its intended function for as long as required but need not be safety grade or single-failure proof.

Further, the staff intends to address compliance with the severe accident requirements defined in Commission policy and 10 CFR Part 52 on a design-specific basis rather than through generic rulemaking as described in SECY-88-248. The staff believes that this approach will minimize scheduling impacts on the individual designs that will ultimately culminate in specific rules through the design certification process.

The staff does not envision any additional requirements to be imposed on the ABWR at this time; however, it will ensure that GE design commitments are sufficient through the detailed review of the design. The staff will inform the Commission during the design certification process if additional requirements are determined necessary for the ABWR design to comply with the Commission's severe accident requirements.

A copy of this paper has been provided to the ACRS so that they might provide their views to the Commission. The staff is scheduled to brief the ABWR Subcommittee on this subject on May 10-11, 1989.

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