January 12, 1990

POLICY ISSUE
(Notation Vote)

SECY-90-016

For: The Commissioners

From: James M. Taylor
Executive Director for Operations

Subject: EVOLUTIONARY LIGHT WATER REACTOR (LWR) CERTIFICATION ISSUES AND THEIR RELATIONSHIP TO CURRENT REGULATORY REQUIREMENTS

Purpose: To present the staff's recommendations concerning proposed departures from current regulations for the evolutionary ALWRs. The staff requests Commission approval of the positions as described in this paper.

Background: In the April 21, July 31, and August 24, 1989 staff requirements memoranda (SRMs), the Commission asked the staff to identify the issues and acceptance criteria used to judge the acceptability of future designs; to identify where the staff proposes to go beyond the regulations or to be less restrictive; and to identify if the Advanced Boiling Water Reactor (ABWR) would meet the Commission's Safety Goal with or without a vent. The Commission asked that these issues be discussed in the context of certification of the ABWR and the other evolutionary advanced light water reactor (ALWR) designs as well as the staff's review of the evolutionary Electric Power Research Institute (EPRI) Requirements Document.

Discussion: Operating experience as well as a number of studies (e.g. PRAs) have identified a number of issues significant to reactor safety. Based on this background the staff has identified the following list of issues as fundamental to agency decisions on the acceptability of evolutionary ALWR designs.

(1) evolutionary LWR public safety goals
(2) source term

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(3) anticipated transients without scram
(4) mid-loop operation
(5) station blackout
(6) fire protection
(7) intersystem LOCA
(8) hydrogen generation and control
(9) core-concrete interaction - ability to cool core debris
(10) high pressure core melt ejection
(11) containment performance
(12) ABWR containment vent design
(13) equipment survivability
(14) operating basis earthquake/safe shutdown earthquake
(15) inservice testing of pumps and valves

The resolutions proposed by EPRI and the LWR vendors, and the staff positions and recommendations regarding each of these issues are discussed in detail in the enclosure. In addition to these issues, each application for a Design Certification will have to propose technical resolutions for those Unresolved Safety Issues and medium- and high-priorty Generic Safety Issues which are identified in NUREG-0933 and technically relevant to the design in accordance with 10 CFR 52.47 and the Severe Accident Policy Statement.

The Commission's approval of, or alternate guidance on, the proposed resolution of these issues is necessary for the staff's continued review of EPRI's ALWR Requirements Document, General Electric's (GE's) ABWR, Westinghouse's RESAR SP/90 and Combustion Engineering's (CE's) System 80+ designs. Approval or guidance is particularly important to the staff's evaluations of the GE ABWR and the EPRI ALWR Requirements Document since these reviews have progressed the furthest. The certification review for CE's System 80+ is just beginning. Westinghouse has indicated that they do not intend to pursue an FDA/certification for the RESAR SP/90 at this time. Additional Commission approval or guidance on significant issues related to certification of CE's System 80+ and other future designs will be discussed with the Commission as part of the development of the licensing review bases (LRB) for these designs. This approach is consistent with recent Commission guidance in an SRM dated December 15, 1989. It should be noted that some of the issues presented in the enclosure are issues proposed by EPRI which they refer to as plant optimization subjects. NRC approval of a plant optimization subject

would result in a resolution that is less restrictive than present regulations, Commission policy, or past licensing practices. For these reasons, the optimization subjects such as hydrogen control, source term, and the relationship between the operating bases earthquake and the safe shutdown earthquake could have a major schedular impact on the evolutionary LWR certification process. The staff has provided the respective applicant's proposed solutions as well as the staff's positions on these issues in the enclosure to provide a comparison, and to provide an indication of the diversity of proposed solutions under consideration by the staff. The staff recommendations identified in this paper have been developed as a result of (1) the staff's reviews of current generation reactor designs and evolutionary ALWRs, (2) consideration of operating experience, including the TMI-2 accident, (3) results of the probabilistic risk assessments (PRAs) of current-generation reactor designs and the evolutionary LWRs, (4) early efforts conducted in support of severe accident rulemaking, and (5) research conducted to address previously identified safety issues. Information related to the issues and staff positions discussed in this paper have previously been provided to the Commission in SECY-89-013, SECY-89-153, SECY-89-228 and SECY-89-341 and have been underlined in the enclosure.

The staff positions recommended in this paper are consistent with those previously taken in the staff's review of the ABWR LRB and in several ABWR-related safety evaluation reports issued to date. The staff believes that, pending detailed staff review, there is a high degree of confidence that the ABWR would meet the positions identified in the enclosure. Therefore, Commission approval of the staff recommendations would close these policy issues for the ABWR and would permit staff review to continue. The recommended positions are also consistent with those identified in the staff's draft safety evaluations related to certain chapters of the EPRI-ALWR Requirements Document. The staff is reviewing severe accident and certification issues addressed in the EPRI-ALWR Requirements Document and the staff's final conclusions are awaiting Commission approval of the positions described in this paper. Approval of the staff recommendations would allow for continuation of the staff review of the EPRI ALWR Requirements Document in accordance with recent Commission guidance.

Conclusions: The staff believes its conclusions and recommendations regarding these matters are in keeping with the Commission's

policy expectation that future designs for nuclear plants will achieve a higher standard of severe accident safety performance.

The staff will inform the Commission during its reviews if additional enhancements to existing requirements, beyond those identified in the enclosure, are determined to be necessary for evolutionary ALWR designs.

Coordination: The Office of General Counsel has reviewed this paper and has no legal objection, but notes that any program for review of new reactor designs which authorizes the NRC to impose requirements beyond those needed to meet current Commission regulations raises the issue that, if the NRC staff can pose additional requirements for certification, other parties should be able to do so as well. The traditional way to avoid such problems is through rulemaking, but as indicated in SECY 89-311, the staff believes that design certification process is a more effective method of resolving severe accident issues than a generic severe accident rule or several individual changes. Further, OGC notes that questions regarding the desirability of additional severe accident mitigation measures still need to be addressed under NEPA, either in the design certification rulemaking and hearing or in some preliminary rulemaking.

A copy of this paper has been provided to the Advisory Committee on Reactor Safeguards.

Recommendations: That the Commission

(1) Approve the staff positions detailed in the enclosure, and

(2) Note that if the staff determines other issues need to be addressed in accordance with Commission guidance, the staff will inform the Commission of its positions on these matters in a timely manner.

(3) Note that the staff, in accordance with the Staff Requirements Memorandum dated August 24, 1989, plans to issue the draft SER on Chapter 5 of the EPRI ALWR Requirements Document.

(4) Note that following Commission resolution of the policy issues discussed in this paper, the staff plans to finalize and reissue the above draft SER.

James M. Taylor
Executive Director
for Operations

Enclosure:
As stated

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Tuesday, January 30, 1990.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Tuesday, January 23, 1990, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional time for analytical review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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EVOLUTIONARY ALWR CERTIFICATION ISSUES

I. General Issues

A. ALWR Public Safety Goal

The EPRI Requirements Document proposes that the evolutionary ALWRs comply with the following public safety goals:

(1) The frequency of core damage will be less than 1.0 X 10E-5 events/per reactor-year, and
(Note: EPRI refers to this as a "quantitative investment protection goal")

(2) Whole body dose at an assumed 0.5 miles site boundary must be less than 25 rem for events whose cumulative frequency exceeds 1 X 10E-6 per reactor-year.

In the Licensing Review Bases (LRB) for the ABWR, GE has committed to meet the following goals:

(1) Demonstrate that the likelihood of core damage will have a mean value of less than one in one hundred thousand reactor years (i.e., 1.0 X 10E-5).

(2) The expected mean frequency of occurrence of offsite doses in excess of 25 rem beyond a half mile radius from the reactor is to be less than once per million reactor years (i.e., 1.0 X 10E-6), considering both internal and external events.

(3) The containment design is to assure that the containment conditional failure probability is less than one in ten when weighted over credible core damage sequences.

The staff is presently reviewing the LRB for the System 80+ design in which CE has proposed goals which are similar to the goals developed in the ABWR LRB. Since Westinghouse has no immediate plans of pursuing certification of RESAR SP/90, work on the LRB is presently not planned. However, similar to GE and CE, Westinghouse has stated, in meetings with the staff and the Advisory Committee on Reactor Safeguards, they are committed to meeting the ALWR public safety goals as well as the goals in the Commission's Safety Goal Policy Statement.

The staff is reviewing the proposed ALWR public safety goals to ensure they are consistent with the Commission's Safety Goal Policy Statement, which proposed both qualitative as well as quantitative safety goals for future reactor designs. The current regulations do not specify requirements in numerical terms of frequency of core damage or large release events. However, the Commission in its Safety Goal Policy Statement, has proposed that the staff examine a general performance guideline that "the overall mean frequency of a large release of radioactivity to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

SECY-89-102 recommended approval by the Commission of the use of the following quantitative objectives in its implementation of the Safety Goal Policy for future standardized plants:

1. The mean core damage frequency target for each design should be less than 1.0 X 10E-5 event per reactor-year, and

2. The overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation where a large release is defined as one that has a potential for causing an offsite early fatality.

The staff concludes that the staff-proposed quantitative safety goals submitted in SECY-89-102 are consistent with the Commission's Safety Goal Policy Statement. Additional Commission guidance on the establishment of quantitative goals and implementation of safety goal policy will assist the staff in its continuing assessment of the evolutionary ALWRs. Although the staff considers the goals defined in SECY-89-102 to be acceptable for evolutionary ALWRs, it should be noted that both the EPRI public safety goal and the ABWR public safety goal are considerably more stringent than the large release guideline defined in SECY-89-102.

Although the staff has indicated it believes the ALWR Public Safety Goal contains meritable goals for the industry to adopt, the staff has not completed its review of this issue and is in the process of reviewing how EPRI implements these goals.

B. Source Term

As noted in SECY 89-341, the staff's methodology for determining compliance with the siting requirements of 10 CFR Part 100 has been based on the 1962 "TID-14844" source term. This methodology, which involves calculation of offsite dose for comparison against Part 100 dose criteria (i.e. criteria for establishing the size of the exclusion area and the low population zone), is widely acknowledged to utilize conservative assumptions. At the time this approach was developed, these conservatisms were considered appropriate and were based on uncertainties associated with accident sequences and equipment performance; and as a means to assure that future plant sites would be essentially equivalent to sites approved up until that time. The conservatisms initially included in the methodology have been essentially retained up to this time.

EPRI has stated that the evolutionary ALWR licensing design-basis requirements as well as design enhancements related to severe accidents should be based on the full body of current knowledge regarding accident source terms. They believe that the evolutionary designs should be evaluated based on a realistic treatment of fission product source terms, including the extensive research that has been done on fission product behavior since TID-14844 was issued, and especially since the Three Mile Island accident in 1979. EPRI's view is that this approach will result in designs which are improved and provide enhanced safety protection. EPRI has identified this as a plant optimization issue.

GE has indicated that the ABWR will meet the offsite dose criteria established in 10 CFR Part 100. However, they propose to utilize updated information such as system performance and reliability information, developed since promulgation of Part 100, to justify some departure from the current methodology for calculating the offsite dose. The ABWR's current design includes a single stand-by gas treatment system (SGTS) charcoal filter bed, and no main steam isolation valve (MSIV) leakage control system (LCS). Previous BWR designs utilized redundant SGTS charcoal filter beds and, since 1976, most have been equipped with a MSIV-LCS. The staff's interpretation of the General Design Criteria (GDCs) would classify filters as active components and require redundancy to permit any dose reduction credit in calculating a Part 100 dose. Since 1976, MSIV-LCSs have been required in most BWRs to meet 10 CFR Part 100 for design bases accidents. Part 100 requires equipment used to mitigate consequences of design basis accidents to be seismically designed (it identifies equipment necessary to mitigate the consequences of accidents whose offsite consequences are comparable to the Part 100 dose guidelines as designed to withstand the vibratory motion of an SSE). Since non-safety grade equipment such as piping downstream of the MSIVs and the condenser are not seismically designed for SSE, credit for these systems has not been accorded in calculating offsite doses for Part 100 purposes. The staff is considering these deviations from the current methodology for demonstrating compliance with Part 100. The staff has concluded, based on current information and experience, that some deviation from current practice, or exemptions from the regulations identified above, may be warranted in the review of evolutionary designs. Presently, the staff believes that no other deviations would be necessary to demonstrate ABWR compliance with Part 100.

The other evolutionary ALWR vendors (Westinghouse and Combustion Engineering) have indicated that their evolutionary designs will comply with 10 CFR Part 100 and that they will work with the NRC and EPRI to utilize more realistic source term information to assess design enhancements related to severe accidents.

As stated in SECY 89-341, the staff is undertaking an examination of the implications of decoupling siting from plant design for future reactors. Under this plan, reactor site characteristics would be reviewed separately from the reactor without utilizing source terms or dose calculations. This would require revision to Part 100 and other regulatory staff practices. The results of such a study will establish appropriate guidelines for any future plant license applications. In the interim, however, the staff recommends that the Commission approve the following approach for evolutionary ALWRs:

- Assure that evolutionary designs meet the requirements of 10 CFR 100
- Consider deviations to current methodology utilized to calculate Part 100 doses on a case-by-case basis utilizing engineering judgement including updated information on source term and equipment reliability.

Such deviations could impact plant design features, therefore, these deviations will be identified in the SERs that will be forwarded to the Commission for its information well in advance of issuance, as directed in the SRM pertaining to SECY 89-311 dated December 15, 1989.

- Do not modify current siting practice, even though it is recognized that such deviations could result in calculated low population zones and exclusion areas which are smaller than those that have been approved for currently operating reactors.

- Continue to interact with EPRI and the evolutionary ALWR vendors to reach agreement on the appropriate use of updated source term information for severe accident performance considerations.

II. Preventative Feature Issues

A. Anticipated Transient Without Scram (ATWS)

The ATWS rule 10CFR 50.62 was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred.

EPRI has indicated that its approach to resolving the ATWS issue is compliance with the ATWS rule. Design requirements beyond those which would be required to meet the rule have not been proposed.

The ABWR design includes a number of features that reduce the risk from an ATWS event. These features include a diverse scram system with both hydraulic and electric run-in capabilities on the control rods, a manually operated standby liquid control system (SLCS), 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 SLCS, GE has concluded that the diverse scram system and enhanced reliability of the reactor protection system negates the need for an automatic SLCS. GE has agreed to provide a reliability analysis in order to support this position. The staff will review the analysis to determine if an exemption from 10 CFR 50.62, to approve manual SLCS, is justified. The staff analysis will be provided in a future safety evaluation report for the ABWR.

Westinghouse has concluded that a diverse scram system is unnecessary for the RESAR SP/90 design due to 1) high reliability of the integrated reactor protection system (IPS), 2) a turbine trip and emergency feedwater actuation that is independent of the IPS, 3) Ability to manually trip the rod control motor generators from the main control board, and 4) a highly negative moderator temperature coefficient. Westinghouse has committed to provide a detailed analysis to demonstrate that the consequences of an ATWS are acceptable at the time an FDA application is submitted.

The CE System 80+ design includes a control-grade Alternate Protection System which provides an alternate reactor trip signal and an alternate feedwater actuation signal separate and diverse from the safety-grade reactor trip system.

The staff believes, notwithstanding the Westinghouse position on diverse scram systems, that all future evolutionary ALWR designs should be required to provide a diverse scram system unless the LWR vendor can demonstrate that the consequences of an ATWS are acceptable. The ATWS rule presently requires a diverse scram system for all (CE, Babcock and Wilcox, and GE) LWR designs except Westinghouse PWRs. It had been determined that previous Westinghouse designs had adequate ATWS capability and backfit could not be justified. The staff believes that evolutionary ALWR designs should provide diverse methods of inserting control rods to mitigate a potential ATWS and to ensure a safe reactor shutdown. The staff considers that diverse scram capability is a worthwhile measure of prevention for all evolutionary ALWRs, especially when incorporated into the initial design. Imposition of a diverse scram system on the Westinghouse design would exceed the Commission's regulations. Therefore, the staff recommends that the Commission approve the staff position that diverse scram systems be provided for evolutionary ALWRs.

B. Mid-Loop Operation

The staff is concerned that decay heat removal capability could be lost when a PWR is shut down for refueling or maintenance and drained to a reduced reactor coolant system (RCS) or "mid-loop" level. For example, a significant problem has been the loss of residual heat removal (RHR) suction due to air-binding of the RHR pumps. This is usually caused by an uncontrolled low loop level and consequent air ingestion into the pump suction.

The EPRI Requirements Document specifies requirements consistent with measures applicable to operating reactors as described by the administrative procedures identified in Generic Letter 88-17, but does not specify design modifications to address the root cause of this event.

Westinghouse has committed to install a vortex breaker at the RHR hot leg connections to significantly reduce air entrainment during mid-loop operation. This feature, in conjunction with other design features of the plant, should greatly reduce concerns over mid-loop operation. CE has indicated that it will address this issue through analysis, consideration of specific design features, and/or operational restrictions. Specific design resolutions for the System 80+ have not been provided. Mid-loop operation is not an issue with the ABWR.

The staff expects improvements in instrumentation in many existing PWRs, but does not require specific modifications to the nuclear steam supply system (NSSS) to correct mid-loop problems. However, the staff believes that physical modifications such as those proposed by Westinghouse, may be necessary to essentially eliminate any concerns with mid-loop operation for future evolutionary pressurized ALWRs. Mid-loop operation is not explicitly covered by current regulations, however imposition of such requirements would exceed current staff licensing practices. Therefore, the staff recommends that the Commission approve the staff position that evolutionary PWR vendors propose design features to ensure high reliability of the shutdown decay heat removal system.

C. 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 EPRI Requirements Document provides for improvements in offsite power reliability, onsite power reliability and capacity, and station blackout coping capability. EPRI is also proposing that a large capacity, diverse alternate ac power source (combustion turbine generator) with the capability to power one complete set of normal safe shutdown loads be included in evolutionary ALWR designs.

The RESAR SP/90 emergency feedwater system includes two ac-independent and two dc-independent turbine-driven pumps. The electrical design includes two full capacity emergency diesel generators. In addition, it includes a backup seal injection pump powered by a small dedicated diesel generator which has enough capacity to also charge the station batteries. Westinghouse believes that this design will provide a 24-hour coping time which is sufficient to eliminate the need for the addition of an installed spare (full capacity) alternate ac power source.

The System 80+ design includes two turbine-driven emergency feedwater pumps and two motor-driven emergency feedwater pumps. The electrical design includes two full capacity emergency diesel generators and a diverse alternate ac power source. This alternate source of ac power is expected to be a control-grade combustion turbine with sufficient capacity and capability to power either one of the electrical divisions. In addition, the plant design has full load rejection capability and the capability to subsequently provide electrical power from the turbine generator. Each of the four safety-related instrument channels has a dedicated battery backup. Class 1E electrical Divisions I and II, which include the two emergency diesel generators are each provided dc power by an assigned pair of these batteries.

The ABWR design includes three independent electrical divisions, each with high-pressure and low-pressure water injection capability, each powered by a full capacity emergency diesel generator, and each division capable of independently shutting down the reactor. 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.

The staff believes that the preferred method of demonstrating compliance with 10 CFR 50.63 is through the installation of a spare (full capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide 1.155, and is capable of powering at least one complete set of

normal safe shutdown loads. Although an alternate ac power source is provided as an acceptable resolution to this issue in 10 CFR 50.63, staff imposition would exceed current Commission regulations. Therefore, the staff recommends that the Commission approve imposition of an alternate ac source for evolutionary ALWRs.

D. Fire Protection

The staff has concluded that fire protection issues that have been raised through operating experience and through the External Events Program must be resolved for evolutionary ALWRs. To minimize fire as a significant contributor to the likelihood of severe accidents for advanced plants, the staff concludes that current NRC guidance must be enhanced. Therefore, the evolutionary ALWR designers must ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible. Because of its physical configuration, the control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. Evolutionary ALWRs must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the evolutionary ALWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. Because the layout of a nuclear plant is design-specific, plant-specific design details will be reviewed by the staff on an individual basis. The staff will require a description of safety-grade provisions for the fire-protection systems to ensure that the remaining shutdown capabilities are protected, as well as demonstration that the design complies with the migration criteria discussed above.

The ALWR Requirements Document indicates that fire protection will be as specified in 10 CFR 50.48 and Appendix R. It states that for equipment in the same general area, a 3-hour fire barrier will be utilized in lieu of physical separation unless it is "impractical or less safe." However, no guidelines are provided in the Requirements Document as to the application of these criteria.

The evolutionary ALWR designers have indicated that their fire protection designs are consistent with the staff's proposed enhancements. GE has provided its ABWR fire protection analysis which is currently under review by the staff.

Appendix R to 10 CFR Part 50 was promulgated for plants that were in operation prior to January 1, 1979. Subsequently, PRAs performed on more than a dozen plants have showed that fire is a significant contributor to core damage. The staff believes that in keeping with the Commission's desire for enhanced safety for evolutionary ALWRs, fire protection requirements should reflect experience from operating reactors and the greater understanding of severe accidents that has been acquired since Appendix R was promulgated. Therefore, the staff recommends the Commission approve the use of the enhanced fire protection position underlined above for evolutionary ALWRs.

E. Intersystem LOCA

Future evolutionary 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.

For both BWRs and PWRs, EPRI states that low-pressure systems which could be overpressurized by the RCS should be designed with sufficient margin to withstand full RCS pressure without structural failure.

For BWRs, pressure isolation valve instrumentation and controls should be provided to (1) prevent opening shutdown cooling connections to the vessel in any loop when the pool suction valve, discharge valve, or spray valves are open in the same loop, (2) prevent opening the shutdown connections to and from the vessel whenever the RCS pressure is above the shutdown range, (3) automatically close shutdown connections when RCS pressure rises above the shutdown range, and (4) prevent operation of shutdown suction valves in the event of a signal that the water level in the reactor is low.

For PWRs, relief valves sized to protect against overpressure transients, should be provided on the RHR system. RHR suction valves should be provided with permissive interlocks to prevent opening if RCS pressure exceeds RHR design pressure.

Westinghouse has indicated that, should the isolation valves of the RESAR SP/90 fail, the design pressure of the piping outside of the containment will be sufficient to withstand primary side pressure or will be vented to the Emergency water storage tank (EWST).

CE has eliminated the low-pressure safety injection system and increased the design pressure of the shutdown cooling system piping in the System 80+ design. With this higher design pressure, the shutdown cooling system is expected to maintain its integrity even when exposed to full reactor coolant system 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 concludes that designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means for resolving this issue. However, the staff believes that for those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should

provide (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room when isolation valve operators are deenergized and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed. Imposition of these requirements exceed Commission regulations and guidance; therefore, the staff recommends that the Commission approve these positions for evolutionary ALWRs.

The staff notes that for some low-pressure systems attached to the RCS, it may not be practical or necessary to provide a higher system ultimate pressure capability for the entire low-pressure connected system. The staff will evaluate these exceptions on a case-by-case basis during specific design certification reviews.

III. Mitigative Feature Issues

A. Hydrogen Generation and Control

The Commission's Severe Accident and Standardization Policy Statements provide 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 of 10 CFR 50.34(f) 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, the regulation requires this system 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 ALWR Requirements Document specifies that containment and combustible gas control systems should be designed to accommodate 75 percent in-vessel zirconium-water reaction of the active fuel cladding, and 13 percent containment uniform hydrogen concentration. It states that 75 percent cladding oxidation is believed to be a conservative upper limit on the amount of hydrogen generated in a degraded-core situation including recovery. EPRI has identified this as an optimization issue.

The RESAR SP/90 design proposes to mitigate the effects of a 100 percent metal-water reaction and to preclude uniform hydrogen concentration from exceeding 10 percent (by volume) through the use of hydrogen igniter and hydrogen recombiner systems.

The System 80+ design proposes to be consistent with the recommendations of the ALWR Requirements Document resulting from staff review. The information will include justifications for the assumed extent of metal-water reaction and the allowable uniform hydrogen concentrations.

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

Aside from the issue of regulatory compliance and applicability, and due to the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, the staff concludes that compliance with the criteria of 10 CFR 50.34(f) remains appropriate for combustible gas control design in ALWRs. Research (discussed in NUREG/CR-4551) indicates that in-vessel hydrogen generation associated with core-damage accidents may range from approximately 40-95 percent active cladding oxidation equivalent. The amount of cladding oxidation is dependent on a variety of parameters related to sequence progression: reactor coolant system pressure, reflood timing and flow rates, as well as core-melt progression phenomena. Thus, a 75-percent-equivalent cladding reaction continues to be viewed as a reasonable design basis for hydrogen generation for severe accidents in which the reactor pressure vessel (RPV) remains intact. However, it is the staff's view that ALWRs should provide protection for hydrogen generation resulting from a wider spectrum of accidents, i.e., full core-melt accidents with RPV failure. In that context, it is also necessary to consider ex-vessel hydrogen generation as a result of core debris reacting with available water or core-concrete interactions. Calculations using the CORCON models indicate that if the core debris is cooled in relatively rapid fashion (1-2 hours), additional hydrogen generation will be less than that equivalent to a 25-percent cladding oxidation reaction. This relatively limited ex-vessel reaction is conditional on the existence of a coolable debris bed and the availability of sufficient water. If extensive core-concrete interaction occurs due to the absence of cavity flooding, more hydrogen generation should be considered. Considering the effects discussed above, the staff concludes that an equivalent 100 percent cladding oxidation reaction is an appropriate deterministic design criteria and a reasonable surrogate for the combination of both in-vessel and ex-vessel hydrogen generation.

Due to the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, it is still the staff's position that, as a minimum, evolutionary ALWRs should be designed to (1) accommodate hydrogen equivalent to 100-percent metal-water reaction of the fuel cladding and (2) limit containment hydrogen concentration to no greater than 10 percent. Furthermore, because hydrogen control is necessary to preclude local concentrations of hydrogen below detonable limits, and given uncertainties in present analytical capabilities, the staff concludes evolutionary ALWRs should provide containment-wide hydrogen control (e.g., igniters, inerting) for severe accidents. Additional advantages of providing hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) includes the lessening of pressure and temperature loadings on the containment and essential equipment. The staff recommends that the Commission approve the staff's position that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary ALWRs.

B. 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, which could contribute to eventual overpressure failure of the containment.

The EPRI Requirements Document contains a number of design features that are intended to mitigate the effects of a molten core. To promote long-term debris coolability, the Requirements Document states that the cavity floor should be sized to provide 0.02 m^2/MWt. The Requirements Document also specifies that the containment should be designed to ensure adequate water supply to the floor and that an alternate means of introducing water into the containment, independent of normal and emergency ac power, should be provided. Passive schemes for providing flooding of the floor areas beneath the vessel are proposed and described in general terms for both BWRs and PWRs. The Requirements Document also states that the steel shell or liner of the containment should be protected from core debris by at least 3 feet of concrete.

Westinghouse indicated that they will comply with the EPRI core-debris dispersal criteria of 0.02 m^2/MWt and that the RESAR SP/90 design will include some method (not yet defined) that would ensure automatic flooding of the lower cavity, using the in-containment refueling water storage tank (RWST), in the event of a severe accident. Westinghouse is currently evaluating alternative designs to ensure compliance with that commitment.

CE has also indicated that the System 80+ design will comply with the EPRI core-debris dispersal criteria of 0.02 m^2/MWt. Also, the in-containment refueling water storage tank will provide a source of water for lower cavity flooding.

The ABWR design has a number of features that the staff generally agrees would mitigate the effects of a molten core. It is designed with a lower drywell flooder and a cavity space sufficient to be able to disperse core debris at an energy level of 0.02 m^2/MWt. 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, ensuring that releases continue to be scrubbed through the suppression pool water. 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 core-concrete interaction.

The staff believes that an acceptable resolution to this issue can be provided by the evolutionary ALWR vendors if their designs

- provide sufficient reactor cavity floor space to enhance debris spreading, and
- provide for quenching debris in the reactor cavity.

Use of these criteria exceed current regulatory practice.

It should be noted that the specific cavity sizing criteria (0.02 m^2/MWt) proposed in the Requirements Document is still under evaluation by the staff. The issue of debris coolability is an area in which there is active ongoing experimental research including relatively large scale testing jointly sponsored by EPRI and NRC. Additionally, without assurance of core debris coolability, the level of protection afforded by a 3-foot thickness of concrete and the issue of vessel pedestal attack (ablation of concrete supporting the reactor vessel by the molten core debris) require further evaluation. The staff will continue to evaluate the issue of core debris coolability and the specific cavity sizing criteria (0.02m^2/MWt) proposed by EPRI as more data and information becomes available. The staff intends to assess the debris flooding schemes proposed by EPRI on a design-specific basis.

The staff recommends the Commission approve exceeding past regulatory practice in resolving this issue. The staff recommends approval of the general criteria, stated above, that evolutionary ALWR designs; 1) provide sufficient reactor cavity floor space to enhance debris spreading, and 2) provide for quenching debris in the reactor cavity. Design specific approaches to resolve this issue will be evaluated by the staff on a case-by-case basis to ensure compliance with these criteria.

C. High Pressure Core Melt Ejection

One potential effect of a severe accident that could potentially result in containment failure is the phenomenon of direct containment heating (DCH). The staff is concerned that this phenomenon might occur from the ejection of molten core debris under high pressure from the reactor vessel resulting in wide dispersal of core debris and extremely rapid addition of energy to the containment atmosphere.

To limit direct containment heating, the ALWR Requirements Document states that the cavity/pedestal/drywell configuration should be designed to preclude entrainment of core debris by gases ejected from a failed reactor vessel. It also states that a safety-grade RCS safety depressurization and vent system (SDVS) will be provided. The staff review has concluded that reactor vessel depressurization capability and cavity design features to entrap ejected core debris constitute an acceptable approach to the issue of high-pressure core melt ejection.

Westinghouse has indicated that the configuration of the cavity of the RESAR SP/90 containment will prevent core debris from entering the upper containment.

In addition, their ac independent depressurization system will reduce the probability of a high-pressure molten-core ejection from the reactor vessel.

CE has indicated the System 80+ design includes an indirect cavity vent path, including a debris collection chamber, (which is configured to de-entrain solid core debris and minimize direct containment heating) and a large floor area to enhance core debris coolability. In addition, the design includes a safety grade depressurization system which minimizes the possibility of high-pressure molten-core ejection.

The ABWR design incorporates a safety grade depressurization system and a suppression pool that surrounds the lower drywell cavity and thereby reduce the risk of high pressure core ejection and would prevent core debris from reaching the containment boundary and breaching its integrity.

The staff concludes that, during a high-pressure core-melt scenario, a depressurization system should provide a rate of RCS depressurization to preclude molten-core ejection and to reduce RCS pressure sufficiently to preclude creep rupture of steam generator tubes. Primary systems of evolutionary ALWRs should have the capability to be depressurized after loss of decay heat removal. In addition, the staff concludes that the ALWR Requirements Document should include a requirement that reactor cavities be arranged in such a manner that high-pressure core debris ejection resulting from vessel failure will not impinge on the containment boundary. The staff concludes that ALWR designs should include a depressurization system and cavity design features to contain ejected core debris. Imposition of these requirements exceed current Commission regulations. The staff recommends that the Commission approve this position for evolutionary ALWRs.

D. Containment Performance

The containment function, i.e., maintenance of a strong leak tight barrier against radioactive release, is faced with distinct challenges as a result of a severe accident. These challenges may be roughly divided into two categories, energetic or rapid energy releases and slower, gradually evolving releases to the closed containment system. Examples of containment loadings that fall into the first category include high-pressure core melt ejection with direct containment heating, hydrogen combustion, and the initial release of stored energy from the reactor coolant system. Slow energy releases to the containment are typified by decay heat and noncondensible gas generation. Engineering practice in containment design calls for passive capability in dealing with energetic energy releases where practicable while long-term energy releases may be controlled by both passive means as well as through active intervention.

In view of the low probability of accidents that would challenge the integrity of the containment, the staff concludes that the probability of failure of the mitigation systems (those systems which can reduce the consequences of a core damage accident), from the onset of core damage to loss of containment integrity resulting in an uncontrollable leakage substantially greater than the design basis leakage, should not exceed approximately 0.1. However, the staff intends

to ensure that the containment can deal with all credible challenges and does not intend to apply this conditional containment failure probability (CCFP) guideline in a manner that could be interpreted to potentially detract from overall safety. The staff will accept a CCFP of 0.1 or a deterministic containment performance goal that offers comparable protection. For this reason, the staff concludes that the following general criterion for containment performance during a severe-accident challenge would be appropriate for the evolutionary ALWRs in place of a CCFP.

> The containment should maintain its role as a reliable leak tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage and that following this 24 hour period the containment should continue to provide a barrier against the uncontrolled release of fission products.

Maintaining containment integrity for a minimum period (e.g. 24 hrs) is based on providing sufficient time for the remaining airborne activity in the containment (principally noble gases and iodine) to decay to a level that would not exceed 10 CFR Part 100 dose guideline values when analyzed realistically, if controlled venting were to occur after that time. During this period, containment integrity should be provided, to the extent practicable, by the passive capability of the containment itself and any related passive design features (e.g., suppression pool). The staff further believes that following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products. However, in keeping with the concept of allowing for intervention in coping with long-term or gradual energy release, the staff believes that after this minimum period, the containment design may utilize controlled, elevated venting to reduce the probability of a catastrophic failure of the containment. Alternatively, a design may utilize diverse containment heat removal systems or rely on the restoration of normal containment heat removal capability if sufficient time is available for major recovery actions (e.g., 48 hours).

EPRI has indicated that the ALWR public safety criteria do not contain explicit criteria for conditional probability of containment failure or other mitigation features since the ALWR Steering Committee believes that such criteria could potentially distort the balance in safety design and inhibit innovative improvements in core protection features. However, EPRI has not yet indicated their position on an alternate containment performance goal.

Westinghouse has not yet committed to a specific containment performance goal for RESAR SP/90 although it is expected that the mitigation features discussed by Westinghouse would lead to a CCFP of less than 0.1 for all credible accident scenarios.

CE expects that the System 80+ design will meet the CCFP goal of 0.1 when weighted over credible core damage sequences given the following assumptions.

-Credible core damage sequences are defined as all core damage event sequences with a frequency of greater than 1.0 X 10E-6 per reactor year. External events which would cause both core damage and concurrently fail the containment and which have a frequency of less that 1.0 X 10E-5 per reactor year will not be considered.

-Containment failure is defined as a post core damage release resulting in a dose greater than 25 rem beyond one-half mile from the reactor.

The ABWR design currently includes a hardened wetwell vent for containment over pressure protection and is committed to a CCFP that is less than 0.1 when weighted over credible core damage sequences. In meetings with the staff, EPRI has stated that they consider a containment vent to be philosophically and institutionally undesirable and potentially unworkable. For additional information see related discussions under ALWR Public Safety Goal and ABWR Containment Vent Design.

Defense in depth, a long standing fundamental principle of reactor safety, results in the concept that multiple barriers should be provided to ensure against any significant release of radioactivity. In its Severe Accident Policy Statement, the Commission indicated that it "... fully expects that vendors engaged in designing new (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs." The Commission reaffirmed this policy in an SRM dated December 15, 1989 relating to SECY-89-311. A defense-in-depth approach reflects an awareness of the need to make conservative safety judgements in the face of uncertainties; in effect, not putting all the eggs in one basket. In that regard, the reactor containment boundary should serve as a reliable barrier against fission product release for credible severe-accident phenomena/challenges. Special effort should be made to eliminate or further reduce the likelihood of a sequence that could bypass the containment. The continued reliance on the traditional principle of containment of fission products following an accident is seen as a logical and prudent approach to addressing reasonable questions which will persist regarding the ability to accurately predict certain aspects of severe accident behavior. In order to ensure balance between prevention and mitigation, some criteria on containment performance are appropriate. Accordingly, a general goal of limiting the conditional containment failure probability to less than 1 in 10 when weighted over credible core-damage sequences would constitute appropriate attention to the defense-in-depth philosophy. Alternatively, a deterministic containment performance goal that provides comparable protection would be appropriate.

Probabilistic risk assessment (PRA) is a very powerful tool that permits systematic integrated assessment of design strengths and weaknesses. However, because very low frequency scenarios (approximately 1.0 X 10E-6 per reactor-year) are being addressed, it is important to recognize the large uncertainties in the quantification of these scenarios. The overall uncertainties in severe accident behavior are driven largely by insufficient data for assessing common-cause failures, difficulty in quantification of the potential for human

errors, and questions about completeness of analyses and uncertainties in phenomenological behavior. For this reason, the staff considers it acceptable to utilize a deterministic containment performance criterion that would provide a level of containment performance comparable to that which could be demonstrated using a probabilistic containment failure goal of 0.1, given a severe accident.

It is recommended that the Commission approve the staff's position to use a CCFP of 0.1 or a deterministic containment performance goal that offers comparable protection in the evaluation of evolutionary ALWRs.

E. ABWR Containment Vent Design

In Amendment 8 of the ABWR SSAR submittal (July 28, 1989) GE submitted a sensitivity analysis of the ABWR PRA to determine the net risk benefit of a vent system. Basically, this system is a containment overpressure relief system and is designed to avoid gross containment failure resulting from postulated slow rising overpressure scenarios that could result from postulated multiple safety system failures. These sensitivity analyses indicate that, with or without a vent system, the ABWR design meets the quantitative health objectives of the Commission's safety goal with a wide margin.

The staff's detailed review of the ABWR risk analyses, including the sensitivity analyses on the vent system, is currently underway. Based on the review to date, the staff believes that the scope of methods and data used in the ABWR PRA are sufficient and do not expect that the ABWR risk to exceed the Commission's quantitative health objectives with or without a vent system.

The staff's safety goal implementation plan also recommended that a subsidiary target related to plant performance be used. This target states that, for future plants, a mean core damage frequency due to internal events and external events be less than 1.0 X 10E-5 per reactor year of operation. The staff's review of Amendment 8 of the ABWR SSAR indicates that the overall core damage frequency from internal events (transients, ATWS events, and postulated LOCAs) and external events (primarily from beyond design basis seismic events and postulated fires) is about 6 X 10E-6 per reactor year. GE has determined that the proposed vent system has negligible impact on the core damage frequency. The staff notes that GE has provided an additional means of decay heat removal (a third train of RHR and an ac-independent water makeup system which relies on the fire water system to supply water to the core and containment sprays in emergency situations) for the ABWR design to reduce the frequency of the sequences involving loss of containment heat removal function, thus reducing the benefit (on core damage frequency only) of the ABWR vent system for these types of accident sequences.

The desirability of venting a BWR containment to mitigate multiple-failure accidents far beyond the design basis has been accepted for some time. Since 1981 the BWR Emergency Procedure Guidelines (EPGs), developed by the BWR Owners Group and approved by the NRC for existing BWRs, have called for venting the containment wetwell air space. GE believes containment

overpressure protection represents a practical and beneficial feature to incorporate in the ABWR. The overpressure protection feature is essentially passive, relatively inexpensive in a new plant, provides insurance against the consequences and financial risks associated with end-of-spectrum accident scenarios, is consistent with the BWR EPGs, and appears to be consistent with the ALWR philosophy of robustness.

GE has established two severe accident goals in the risk analyses submitted to the staff. These goals were defined in the ABWR LRB. The first goal states that the frequency of a severe accident release resulting in a whole body dose of 25 rem beyond one-half mile from the reactor should not exceed 1 X 10E-6 per reactor-year. This design goal is basically the same as the EPRI ALWR design goal. The second goal defined in the ABWR LRB states that the conditional containment failure probability should be less than one in ten (CCFP 0.1) when weighted over credible core damage sequences. The staff and GE agree that the definition of containment failure is an uncontrollable leakage substantially greater than the design basis resulting from loss of containment integrity following the onset of severe core damage. The ABWR design with the vent system is expected to meet the above goals; however, staff review in this area is not yet complete.

GE has performed an analysis utilizing this definition of containment failure to determine if the ABWR meets the CCFP goal of 0.1. The analysis indicates that the CCFP for the ABWR design, without a vent system, is equal to approximately 0.5 and does not meet the 0.1 goal, however with a vent system, the CCFP equals approximately 0.06.

Based upon the preliminary review of the ABWR severe accident design, the staff has determined that, as far as overall risk impact is concerned, the GE ABWR public safety goal is significantly more stringent than the Commission's quantitative health objectives. Also, the staff concludes that the public safety goal proposed by GE for the ABWR design is more stringent than the "large release guideline" as defined in the staff's proposed safety goal implementation plan. Therefore, based on the apparent enhanced level of safety provided by the ABWR's severe accident design features, which include the over pressure protection system, the staff recommends the Commission approve its use in the ABWR design certification process.

F. Equipment Survivability

With regard to the Commission's request concerning "The measures to ensure that systems and equipment required only to mitigate severe accidents are available to perform their intended function (e.g., environmental qualifications)," the staff believes that features provided for severe-accident protection (prevention and mitigation) only (not required for design basis accidents) need not be subject to (a) the 10 CFR 50.49 environmental qualification requirements, (b) all aspects of 10 CFR Part 50, Appendix B quality assurance requirements, or (c) 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgment is that the staff does not believe that severe core damage accidents should be design basis accidents (DBA) in the traditional sense that DBAs have been treated in the past.

Notwithstanding that judgment, however, mitigation features must be designed so there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In instances where safety related equipment, (which is provided for design bases accidents) is relied upon to cope with severe accidents situations; there should also be a high confidence that this equipment will survive severe accident conditions for the period that is needed to perform its intended function. However it is not necessary for redundant trains to be qualified to meet this goal.

During the review of a specific ALWR design the credible severe accident scenarios, the equipment needed to perform mitigative functions, and the conditions under which the mitigative systems must function, will be identified. Equipment survivability expectations under severe accident conditions should include consideration of the circumstances of applicable initiating events (e.g., station blackout, earthquakes) and the environment (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function. The required system performance criteria will be based on the results of these design-specific reviews. In addition, the staff concludes that severe-accident mitigation equipment for evolutionary ALWRs should be capable of being powered from an alternate power supply as well as from the normal Class 1E onsite systems. Appendices A and B to Regulatory Guide 1.155, "Station Blackout," provide guidance on the type of quality assurance activities and specifications which the staff concludes are appropriate for equipment utilized to prevent and mitigate the consequences of severe accidents.

The staff requests that the Commission approve the staff position that features provided only for severe-accident protection need not be subject to the 10 CFR 50.49 environmental qualification requirements, 10 CFR Part 50, Appendix B quality assurance requirements, and 10 CFR Part 50, Appendix A redundancy/diversity requirements.

IV. Non-Severe Accident Issues

The following issues, which are not normally considered through PRA analysis or not considered as severe accident issues for the evolutionary ALWRs, are brought to the Commission's attention because either the staff's positions or the vendor requests differ from past practices.

A. Operating Bases Earthquake(OBE)/Safe Shutdown Earthquake(SSE)

Presently, 10 CFR Part 100 requires that the magnitude of the OBE be at least one-half that of the SSE. It has been an industry wide experience that such a requirement leads to a design that is governed by the OBE requirements and produces unnecessary and inconsistent margins for the SSE loading. This requirement was included in the regulation when the staff did not have substantial experience with the seismic resistance of plants that incorporated OBE design at half the SSE value. Since then a number of research programs have been conducted including a large industry effort on testing and observation of actual earthquake experience of industrial facilities; consequently, the NRC funded Piping Review Committee has concluded that the OBE at existing plants are too high, therefore, controlling the design of some safety systems, and recommended that the OBE be decoupled from the SSE. Certain interim measures,

such as allowing somewhat higher damping values for piping analysis, have been taken to partially implement the Piping Review Committee recommendations (NUREG 1061, 1984). But the complete implementation of the recommendations would involve a revision of 10 CFR Part 100, Appendix A. Because of higher priority work, the effort on revision of this regulation has been postponed. It should be noted that the Commission has, in certain site specific cases, previously approved OBEs of less than one-half the SSE.

EPRI has requested that NRC regulations be changed to reduce the magnitude of the OBE relative to the SSE as a basis for the design. All evolutionary ALWR vendors agree with the request. GE has stated that they agree with EPRI in principle, however, the ABWR design uses an OBE that is one-half the SSE; therefore, this is a non-issue for the ABWR. EPRI has identified this as an optimization issue.

The staff agrees that the OBE should not control the design of safety systems. However, a staff position on this issue to be applied generically to all future designs has not yet been fully developed. For the evolutionary reactors, the staff will consider requests to decouple the OBE from the SSE on a design-specific basis. Such a decoupling would require an exemption to the Commission's regulations, therefore the staff recommends that the Commission approve this design-specific relief approach.

B. Inservice Testing of Pumps and Valves

The ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" has been used to establish past testing requirements for ASME Code Class 1, 2, and 3 safety-related pumps and valves. These requirements provide certain information on the operational readiness of the components, but in general, do not necessarily verify the capability of the components to perform their intended safety function. It is the staff's judgement that the Code does not assure the necessary level of component operability that is desired for the evolutionary LWR designs. The staff believes that the following aspects of pump and valve testing and inspection are necessary to provide an adequate level of assurance of operability. The following provisions should be applied to all safety related pumps and valves and not limited to ASME Code Class 1, 2, and 3 components.

-Piping design should incorporate provisions for full flow testing (maximum design flow) of pumps, and check valves.

-Designs should incorporate provisions to test motor operated valves under design basis differential pressure.

-Check valve testing should incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics.

-A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation which cannot be detected through the use of advanced non-intrusive techniques.

The staff has informed EPRI and the evolutionary ALWR designers of its concerns. No position has yet been expressed by these groups. Imposition of these procedures would exceed past licensing guidance; therefore, the staff recommends that the Commission approve these provisions for evolutionary ALWRs.