UNITED STATES NUCLEAR REGULATORY COMMISSION

POLICY ISSUE
(Notation Vote)

April 2, 1993 SECY-93-087

FOR: The Commissioners

FROM: James M. Taylor
Executive Director for Operations

SUBJECT: POLICY, TECHNICAL, AND LICENSING ISSUES PERTAINING TO EVOLUTIONARY AND ADVANCED LIGHT-WATER REACTOR (ALWR) DESIGNS

PURPOSE:

To present the Commission with recommended positions pertaining to evolutionary and passive light-water reactor (LWR) design certification policy issues and to request that the Commission approve the underlined staff positions presented in this paper.

SUMMARY:

In Enclosure 1, the Nuclear Regulatory Commission staff discusses 42 technical and policy issues pertaining to either evolutionary LWRs, passive LWRs, or both. The staff previously identified these issues in the draft Commission papers, "Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements," dated February 20, 1992, and "Design Certification Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," dated June 25, 1992. After considering the Advisory Committee on Reactor Safeguards (ACRS), industry, and vendor comments, the staff has reached a final position on many of the issues and the staff has underlined the positions for which it is requesting the Commission's approval. The staff also discusses other issues,

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NOTE: TO BE MADE PUBLICLY AVAILABLE IN 3 WORKING DAYS FROM THE DATE OF THIS PAPER which it concludes may be of interest to the Commission. For these issues, the staff will update the Commission, if appropriate, after reaching a final staff position or as warranted by new, substantive information on the issues.

BACKGROUND:

The staff has forwarded several policy papers[1] to the Commission proposing resolutions for policy matters and major technical issues concerning both evolutionary and advanced LWR designs. Two draft Commission papers, cited in the above summary, were released to the public after they were forwarded to the Commission. Those papers supported dialogue between the NRC staff, the ACRS, the Electric Power Research Institute (EPRI), the vendors, and other industry representatives which was focused on resolving the policy and technical issues.

The staff has considered the comments received and reached a final position on many of the issues listed in the draft Commission papers. The staff has finalized its position on 20 issues and requests that the Commission approve the positions recommended in this paper. The staff has also determined that 9 issues discussed in the draft Commission papers are not policy issues and the staff does not anticipate any future interaction with the Commission concerning these issues. For these 9 issues and for the 13 issues which the staff will discuss its final position in future Commission papers, the staff discussions are for the Commission's information only.

DISCUSSION:

Enclosure 1 discusses the staff's positions and the current regulatory requirement or interpretation, as well as comments received from the ACRS, the industry, and the vendors regarding 42 technical and policy issues pertaining to evolutionary LWR designs, passive LWR designs, or both. Where appropriate, the staff has included a detailed discussion of the basis for its position on each issue. The staff has also underlined the positions for which it is requesting the Commission's approval.

Enclosure 1 is divided into three sections. Section I discusses issues previously identified to the Commission in SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." Section II discusses other evolutionary and passive LWR design issues and Section III discusses issues which are applicable only to passive LWR designs. Preliminary analysis and recommendations for issues discussed in Sections II and III were previously transmitted to the Commission in draft Commission papers "Issues Pertaining to Evolutionary and Passive

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[1] Enclosure 3 lists the papers that the staff has forwarded to the Commission regarding policy issues identified for evolutionary and passive advanced light-water reactors. The staff references applicable documents throughout this paper.

Light-Water Reactors and Their Relationship to Current Regulatory Requirements," dated February 20, 1992, and "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," dated June 25, 1992.

Enclosure 2 cross-references issues pertaining to evolutionary, and passive LWR designs with the Commission papers in which the staff has addressed each issue. Enclosure 3 lists Commission papers related to ALWR designs.

The staff developed the recommendations in this paper after:

(1) reviewing current operating reactor designs, evolutionary designs, and passive ALWR design information which was available;

(2) considering operating experience;

(3) considering insights from the available results of the probabilistic risk assessments (PRAs) of LWRs and ALWRs;

(4) considering the Commission's guidance on issues resolved for the evolutionary ALWRs;

(5) completing the draft safety evaluation report for the EPRI Utility Requirements Document (URD) for passive ALWR designs;

(6) completing the final safety evaluation report for the EPRI URD for evolutionary ALWR designs; and,

(7) considering EPRI, ACRS, and industry comments on these issues.

The staff concludes that the positions discussed in Enclosure 1 are fundamental to the Agency's decisions on the acceptability of the evolutionary and passive LWR designs. As discussed in SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light-Water Reactor (LWR) Designs," the staff proposes to implement final positions on these matters as approved by the Commission through individual design certifications and generic rulemaking, as appropriate.

The staff plans to forward to the Commission and solicit ACRS and industry comments on at least two additional Commission papers which will discuss issues relating to (1) the regulatory treatment of nonsafety systems in passive designs and (2) use of a physically based source term.

CONCLUSIONS:

The staff requests that the Commission approve the recommended positions for issues pertaining to evolutionary LWR designs. Such approval would enable the staff to proceed with the final design approval and the design certification review of GE Nuclear Energy's (GE) Advanced Boiling Water Reactor and Asea Brown Boveri-Combustion Engineering's (ABB-CE) System 80+ LWR designs.

The staff also requests that the Commission approve the proposed positions for issues pertaining to the passive designs. This will enable the staff to proceed more effectively with its review of Westinghouse's AP600 and GE's Simplified Boiling Water Reactor ALWR designs.

COORDINATION:

The Office of General Counsel (OGC) has reviewed this paper and has no legal objection. OGC notes that Commission approval would be tentative, subject to further review in design certification rulemakings, and that communications with vendors and EPRI regarding these Commission positions should state this fact.

RECOMMENDATIONS:

The staff recommends that the Commission

(1) Approve the positions underlined in Enclosure 1.

(2) Note that the staff is still considering other policy issues and it will seek the Commission's approval of its positions in the future.

James H. Snezek /for

James M. Taylor
Executive Director
for Operations

Enclosures:
1. Policy Issues Analysis and Recommendations
2. ALWR Issue Cross-Reference Matrix
3. Commission Papers Applicable to ALWRs.

Commissioners' comments or consent should be provided directly to the Office of the Secretary by COB Monday, April 19, 1993.

Commission Staff Office comments, if any, should be submitted to the Commissioners NLT Monday, April 12, 1993, with an information copy to the Office of the Secretary. If the paper is of such a nature that it requires additional review and comment, the Commissioners and the Secretariat should be apprised of when comments may be expected.

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POLICY ISSUES ANALYSIS
AND RECOMMENDATIONS

This enclosure is divided into three sections. Section I discusses issues previously identified to the Commission in SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." Section II discusses other evolutionary and passive LWR design issues and Section III discusses issues which are applicable only to passive LWR designs. Preliminary analysis and recommendations for issues discussed in Sections II and III were previously transmitted to the Commission in draft Commission papers "Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements," dated February 20, 1992, and "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," dated June 25, 1992.

The staff has reviewed comments from industry and the Advisory Committee on Reactor Safeguards (ACRS) regarding the staff positions discussed in SECY-90-016 and the preliminary staff positions discussed in the two draft Commission papers. The staff has evaluated these comments, continued to dialogue with the industry and the ACRS and, where appropriate, revised our discussions to address industry and ACRS comments. The staff requests that the Commission review the discussions contained in this enclosure and approve the underlined staff positions.

I. SECY-90-016 Issues

A. Use of a Physically Based Source Term

This section provides an overview of the current source term status and discusses source term policy issues previously identified in SECY-90-016. A complete source term discussion pertaining to advanced light-water reactor (ALWR) design will be presented in the forthcoming Commission paper on source term.

For approximately 30 years, the Nuclear Regulatory Commission (NRC) staff has been using the reactor accident source term guidelines contained in Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites (March 1962)," to evaluate design-basis-accident (DBA) analyses.

In SECY-90-016, the staff discussed the methodology for determining compliance with the siting requirements of 10 CFR Part 100 using TID-14844. The staff noted that the assumptions in the TID-14844 methodology considered the uncertainties associated with accident sequences and equipment performance and were intended to ensure that future plant sites would provide sufficient safety margins. This methodology has remained essentially unchanged and many of its original assumptions are considered outdated.

In SECY-90-016, the staff recommended that the Commission approve the following approach for evolutionary ALWRs:

1. Ensure that evolutionary designs meet the requirements of 10 CFR Part 100.

2. Consider deviations from current methodology used to calculate 10 CFR Part 100 doses on a case-by-case basis using engineering judgement including updated information on source term and equipment reliability.

3. Do not modify current siting practice.

4. Continue to interact with the Electric Power Research Institute (EPRI) and the evolutionary ALWR vendors to reach agreement on the appropriate use of updated source term information for severe accident performance considerations.

In its staff requirements memorandum (SRM) of June 26, 1990, the Commission approved the staff's approach in determining the source term for the evolutionary designs. The Commission also directed the staff to modify regulations, regulatory practices, and the review process, as appropriate, to reflect information resulting from source term research.

As a result of this guidance, the NRC staff has developed a new source term based on calculations performed using the source term code package for individual accident sequences selected in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," dated December 1990. The proposed new source term is discussed in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" and was issued in July 1992 as a draft report for public comment. NUREG-1465 discusses radionuclide release fractions, release timing, and chemical forms of fission products that would be released into containment based upon a range of core melt accident scenarios, including failure of the reactor vessel and subsequent molten core-concrete interactions.

The new source term document was not intended and does not provide a specific methodology or implementation guidelines for the staff's licensing review of advanced light-water reactors. Needed information includes quantification of parameters related to the removal or reduction of fission products within containment via the use of engineered safety features such as sprays and filters, and passive processes such as aerosol deposition and plateout. The NRC staff is currently working with Sandia National Laboratory to evaluate fission product removal mechanisms within containment and quantify associated releases to the environment.

EPRI has proposed source terms which are based on bounding severe reactor accidents. EPRI provided the staff with technical justification for their new source terms in correspondence dated October 18, 1990, and February 12, 1991, entitled "Licensing Design Basis Source Term Update for the Evolutionary ALWR," and "Passive ALWR Source Term," respectively. The EPRI-proposed source terms are based on single, enveloping values for bounding severe reactor accident sequences, using release data obtained from (1) the Severe Fuel Damage Tests at the Power Burst Facility, (2) the Loss of Fluid Test (LOFT) source term measurements, and 3) data from the TMI-2 post accident examination.

For the evolutionary designs, the EPRI source terms address fission product fuel release magnitude, the fission product release timing, the chemical form of iodine, the retention of aerosol in the reactor coolant and the use of the suppression pool and containment sprays for removal of aerosol and soluble gases. For the passive designs, EPRI proposes that the source terms also consider passive mitigation functions and systems such as steam condensation-driven aerosol removal, main steam isolation valve leakage control, and secondary building fission product leakage control.

The staff compared the draft source terms in NUREG-1465 with EPRI evolutionary and passive LWR source terms specifically focusing on the accident severity selected, the nature of the release phases, and the timing and magnitude of the important nuclides released. The staff concludes that there is, in general, good agreement between the EPRI-proposed source term for the passive plants and the staff's proposed source term. However, the staff noted that the staff values for the magnitude of low volatility radionuclides were about an order of magnitude higher than those proposed by EPRI. These differences appear to be attributed to EPRI's assumption that little or no fission products will be released to the containment atmosphere from ex-vessel debris following core-concrete interaction. According to EPRI, this is due to the ability of the passive ALWR to provide ample coolant to the reactor cavity/ lower drywell prior to or immediately upon vessel penetration. The NRC staff model does not make this assumption.

GE Nuclear Energy (GE) demonstrated in its standard safety analysis report that the Advanced Boiling Water Reactor (ABWR) design will meet the offsite dose reference values set forth in 10 CFR Part 100 using the current TID-14844 source term. The staff has reviewed the ABWR design and performed an independent analysis of the radiological consequences resulting from a postulated DBA and concluded in the draft final safety evaluation that the ABWR design will meet the dose reference values set forth in 10 CFR Part 100.

Asea Brown Boveri-Combustion Engineering, Inc. (ABB-CE) initially proposed the use of the TID-14844 source term and the existing analyses in the ABB-CE System 80+ standard safety analysis report are based on the TID-14844 source term. However, ABB-CE is evaluating the possibility of adopting the NUREG-1465 source term.

In its AP600 design, Westinghouse proposed the same accident source term as proposed by EPRI for passive plants. In its simplified boiling water reactor (SBWR) design, GE proposed the same accident source term as published in NUREG-1465. The staff continues to interact with the passive plant vendors to resolve source-term related issues.

The staff will continue its approach for review of source terms proposed by EPRI and evolutionary and passive plant vendors (as approved in the June 26, 1990, SRM) without waiting for final agency adoption of NUREG-1465. The staff evaluations of ALWR submittals will utilize the current source term research insights regarding fission product release into containment. In determining the effects of removal mechanisms such as sprays, filters, plateout and aerosol deposition, the staff will utilize engineering judgement and best estimates for the applicable parameters.

The staff is considering several source term related design certification issues. The staff previously identified several source term related policy issues involving control room habitability, radionuclide attenuation, and containment bypass in two draft Commission papers. In addition, several new source term related policy issues appear to be developing which may concern (1) dose assessments for ex-vessel releases which result from a severe accident scenario (inclusion of release from core-concrete interaction), (2) assessment of safety-related equipment qualification and (3) assessment of post-accident sampling capabilities and vital area access provisions. The staff's proposed resolution of these issues will be discussed in a separate Commission paper which will discuss source term related issues.

B. Anticipated Transient Without Scram

As discussed in SECY-90-016, the ATWS Rule (10 CFR 50.62) was promulgated to reduce the probability of an anticipated transient without scram (ATWS) and to enhance mitigation capability if such an event occurred. The staff recommended that the Commission approve its position that diverse scram systems should be provided for evolutionary ALWRs. In addition, the staff indicated that GE would perform a reliability analysis to determine whether they could justify manual operation of the standby liquid control system (SLCS), in lieu of automatic operation as required by 10 CFR 50.62, in the event of an ATWS.

In its SRM of June 26, 1990, the Commission approved the staff's position. However, the Commission directed that the staff should accept an applicant's alternative to the diverse scram system, if the applicant can demonstrate that the consequences of an ATWS are acceptable.

The ABWR design includes a number of features that reduce the risks associated with an ATWS event. These features include a diverse scram system with both hydraulic and electric run-in capabilities on the control rods, a SLCS, and a recirculation pump trip capability. In its letter dated October 9, 1991, GE indicated that it will automate the SLCS and they subsequently provided this information in amendment 20 to the SSAR. In addition, the scram discharge volume has been removed from the ABWR design, eliminating some of the potential ATWS problems associated with the older boiling water reactor (BWR) designs.

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

The staff has evaluated the GE ABWR and the ABB-CE System 80+ evolutionary LWR designs and concludes that the designs adhere to the Commission's guidance regarding diverse scram systems.

In its letter of December 6, 1991, EPRI stated that it has determined that automatic actuation of the SLCS was appropriate, and that it was modifying the requirements document for evolutionary designs to reflect that position. EPRI no longer considers this to be a plant optimization subject.

In its requirements documents, EPRI provides design requirements that are consistent with the staff's position on ATWS discussed in SECY-90-016, as modified by the Commission's SRM of June 26, 1990. In its letter of May 5, 1992, EPRI indicated that its approach to resolving the ATWS issue, for both evolutionary and passive designs, is compliance with the ATWS Rule. EPRI has not proposed design requirements beyond those required to meet the rule.

The passive ALWR vendors have indicated that their designs will comply with the requirements document for passive designs, and the staff is evaluating passive designs to ensure compliance with Commission regulations and guidance regarding ATWS. The staff considers this policy issue resolved.

C. Mid-Loop Operation

In SECY-90-016, the staff stated its concern that decay heat removal capability could be unavailable when a pressurized-water reactor (PWR) is shut down for refueling or maintenance and drained to a reduced reactor coolant system (RCS) or "mid-loop" level. The staff recommended that the Commission approve its position that evolutionary PWR vendors must propose design features to ensure high reliability of the shutdown decay heat removal system.

In its letter of April 26, 1990, the ACRS recommended that the staff consider four additional requirements to resolve this issue. In its memorandum of April 27, 1990, the staff indicated that it would ensure that the four ACRS recommendations would be considered during the review of evolutionary PWR designs.

In its SRM of June 26, 1990, the Commission approved the staff's position and endorsed consideration of the four additional ACRS requirements for mid-loop operation. However, the staff remained concerned that the vendors and EPRI have not adequately evaluated the overall question regarding the vulnerability of the ALWRs during shutdown and low-power operation. This issue was discussed with regard to evolutionary designs in a memorandum to the

Commission dated September 5, 1990. The staff requested that ALWR vendors and EPRI assess shutdown and low-power risk, identifying design-specific vulnerabilities and weaknesses and documenting their consideration and incorporation of design features that minimize such vulnerabilities.

In its letter of December 16, 1991, EPRI submitted proposed changes to the requirements document to address this issue for the evolutionary designs. In its letter of May 5, 1992, EPRI stated that the Passive Requirements Document specifies extensive deterministic requirements to address known shutdown risks based on industry experience. EPRI also requires probabilistic and operational shutdown risk evaluations and analysis and has submitted a revision to the requirements documents to address additional requirements resulting from its review of NUREG-1410, "Loss of Vital AC Power and the Residual Heat Removal System During Midloop Operations at Vogtle Unit 2 on March 20, 1990," and NUREG 1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States."

The EPRI requirements document and evolutionary PWR designers have provided features to address the issue of mid-loop operation. The staff has reviewed the features for the ABB-CE System 80+ design and concludes that this design adequately implements the Commission's guidance on this issue.

The staff concludes that passive plants must also have a reliable means of maintaining decay heat removal capability during all phases of shutdown activities, including refueling and maintenance, and will evaluate the adequacy of designs during its review. The staff does not consider this issue to be a policy matter, but rather an element of its normal review. Therefore, the staff considers this policy issue resolved.

D. Station Blackout

As discussed in SECY-90-016, 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 on-site) is unavailable. The staff concluded 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. This power source should be consistent with the guidance in RG 1.155, and should be capable of powering at least one complete set of normal shutdown loads. The staff recommended that the Commission approve its position mandating an alternative ac power source for evolutionary ALWRs. In its SRM of June 26, 1990, the Commission approved the staff's position.

In addition to other design features to address the issue of station blackout, the EPRI requirements document for evolutionary designs and the evolutionary ALWR vendors have provided for a large-capacity, alternative ac power source (combustion turbine generator) with the capability to power one complete set of normal safe-shutdown loads. The staff concludes that the EPRI proposal that calls for a "combustion turbine unit" which EPRI has concluded would "achieve diversity of power sources and maximize the overall reliability of the on-site standby ac power supply system" meets the intent of the Commission guidance on this issue. The staff is still in the process of evaluating the ALWR vendor submittals to ensure acceptable implementation of the Commission's guidance on this issue, but does not expect any related policy matters to result from its review.

Because the passive ALWR designs do not rely on active systems for safe shutdown following an event, EPRI and the passive plant designers have indicated that both safety-related diesel generators and an alternate ac power source should not be required. However, the staff believes that the diesel generators included in passive LWR designs may require some regulatory treatment.

The staff is still evaluating this issue for the passive plant designs. The staff's proposed resolution of this issue will be discussed in a separate Commission paper which will discuss the regulatory treatment of nonsafety systems in passive plant designs.

E. Fire Protection

As discussed in SECY-90-016, the staff recommended that current NRC guidance to resolve fire protection issues should be enhanced to minimize fire as a significant contributor to the likelihood of severe accidents for advanced plants. The staff proposed to require that 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 ALWR designers 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, 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.

In its letter of April 26, 1990, the ACRS recommended that the staff consider additional matters in its evaluation of the fire protection designs such as redundant train separation. In its response to the ACRS letter dated April 27, 1990, the staff stated that the proposed requirement to consider the effects of smoke, heat, and fire suppressant migration may warrant separate heating, ventilation, and air conditioning (HVAC) systems, but that other options may be available to the designer. In its SRM of June 26, 1990, the Commission approved the staff's position, as supplemented by the staff's response of April 27, 1990.

In a letter dated May 5, 1992, EPRI stated, "The ALWR program requirements have been specified to provide for three-hour fire barriers between redundant safety systems and to prevent the migration of smoke and hot gases between compartments containing redundant safety systems by requiring design of separate HVAC systems to serve redundant trains of safety equipment. Exceptions to these requirements are the main control room and reactor containment."

In its letter dated August 17, 1992, ACRS stated, "Except for the concern with shared HVAC, we support the staff recommendation that the passive plants should be reviewed against the enhanced fire protection criteria approved in the Commission's SRM." The ACRS concern with shared HVAC relates to the need for adequate isolation of such systems during certain disruptive events (such as fires, floods, or pipe breaks). ACRS stated:

> If the isolation is not adequate, the HVAC arrangement may become a pathway whereby effluents from the event are conducted to locations where required safe-shutdown equipment is located. This is not a concern if either (1) the HVAC isolation provisions are able to withstand the event consequences (e.g., pipe whip, jet impingement, static and dynamic pressure, and elevated temperature) during and after closure with consideration of single active component failures and acceptable leakage, or (2) the safe shutdown equipment is qualified for the environmental exposure resulting from a release of the adverse environment at any credible location along the HVAC pathway, such as duct openings or blowout locations.

The staff maintains that the proposed requirement to consider the effects of smoke, heat, and fire suppressant migration may warrant separate HVAC systems, but that other options may be available to the designer.

The EPRI Requirements Document and the evolutionary ALWR designers have indicated that their fire protection designs are consistent with the staff's proposed enhancements. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's guidance on this issue, and does not expect any related policy matters to result from its review.

EPRI has specified requirements for the passive designs similar to those for the evolutionary designs. The passive ALWR vendors have indicated that their designs will comply with the requirements document for passive designs.

The staff has not identified any unique features of the passive designs that would preclude the staff's conclusion that these designs should also be evaluated against the enhanced fire protection criteria. Therefore, the staff recommends that the Commission approve the position that the passive plants should also be reviewed against the enhanced fire protection criteria approved in the Commission's SRM of June 26, 1990.

F. Intersystem Loss-of-Coolant Accident

In SECY-90-016, the staff recommended that evolutionary ALWR designers should 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 RCS to withstand the full RCS pressure. The staff further recommended that systems that have not been designed to full RCS pressure should include:

- the capability for leak testing of the pressure isolation valves;

- valve position indication that is available in the control room when isolation valve operators are deenergized; and,

- high-pressure alarms to warn control room operators when rising reactor coolant (RC) pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

In its SRM of June 26, 1990, the Commission approved the staff's position on intersystem LOCA, provided that all elements of the low-pressure system are considered (including instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

The EPRI requirements document and the evolutionary ALWR designers have indicated that their submittals are consistent with the approved resolution to this issue. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's guidance on this issue, but does not expect any related policy matters to result from its review.

EPRI has specified requirements for the passive designs similar to those for the evolutionary designs. The passive ALWR vendors have indicated that their designs will comply with the applicable requirements document.

In its letter of May 5, 1992, EPRI stated that in order to ensure that the guidance of SECY-90-016 is applied to all systems and subsystems connecting to the RCS, the Utility Requirements Document (URD) was revised to include the following specific requirements:

- All systems and subsystems connected to the RCS which extended outside the primary containment boundary must be designed to the extent practicable to an ultimate rupture strength (URS) of at least equal to full RCS pressure.

- The designer must determine by evaluation that, for interfacing systems or subsystems which do not meet the full RCS URS requirement, the degree and quality of isolation or reduced severity of the potential pressure challenges are sufficient to preclude an intersystem LOCA.

- Additional testing and control room alarm capabilities must be implemented to help reduce the probability of an intersystem LOCA.

In general, the staff has found that these requirements are consistent with the staff position. However, as stated in the draft safety evaluation report (SER) on the passive URD, it will be necessary for the plant designer to demonstrate that any interfacing system for which the URS is not at least equal to full RCS pressure could not practically be designed to meet such a criterion. The degree of isolation or number of barriers (for example, three isolation valves) is not sufficient justification for using low-pressure components that can practically be designed to the full RCS URS criterion. In addition, piping runs should be designed to meet the full RCS URS criterion, as should all associated elements (such as flanges, connectors, packing, valve stem seals, pump seals, heat exchanger tubes, valve bonnets, and RCS drain and vent lines). The plant designer should make every effort to reduce the level of pressure challenge to all systems and subsystems connected to the RCS.

The staff has not identified any unique features of the passive plant designs that would preclude the staff's conclusion that these designs should also be evaluated against the staff's previous recommendation. Therefore, the staff recommends that the Commission approve the position that the passive plants should also be reviewed for compliance with the intersystem LOCA criteria approved in the Commission's SRM of June 26, 1990.

G. Hydrogen Control

Containments are required to be designed for control of hydrogen generation following an accident and 10 CFR 52.47(a)(ii) requires all applicants for design certification to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements. For example, 10 CFR 50.34(f)(2)(ix), *Additional TMI-related requirements*, requires a hydrogen control system that can safely accommodate hydrogen generated by the equivalent of a 100-percent fuel-clad metal water reaction. The system must also ensure that uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent (by volume), or that the post-accident atmosphere will not support hydrogen combustion.

Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, the staff recommended in SECY-90-016 that evolutionary ALWRs should be designed, as a minimum, to satisfy the following criteria:

- accommodate hydrogen generation equivalent to a 100-percent metal-water reaction of the fuel cladding;

- limit containment hydrogen concentration to no greater than 10 percent; and,

- provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents.

The staff recommended that the Commission approve its position that the requirements of 10 CFR 50.34(f)(2)(ix) should remain unchanged for evolutionary ALWRs. In its SRM of June 26, 1990, the Commission approved the staff's position.

In its letter of December 6, 1991, EPRI stated that its requirements document for evolutionary designs will be modified to fully comply with the above positions. The ABWR design meets the requirements of 10 CFR 50.34(f)(2)(ix) by using, in conjunction with other systems, a nitrogen-inerted containment atmosphere. The ABB-CE System 80+ has a hydrogen mitigation system consisting of igniters to accommodate the hydrogen production from a 100-percent fuel-clad metal-water reaction and maintain the average containment hydrogen concentration below 10 percent. The staff is in the process of evaluating these submittals to ensure acceptable implementation of the Commission's guidance on this issue but does not expect any related policy matters to result from its review.

In a letter dated May 5, 1992, EPRI indicated that the Requirements Document for evolutionary plants had been revised to require use of a containment-wide hydrogen control system that meets the requirements of 10 CFR 50.34(f). In addition, EPRI indicated that changes were being developed to require a hydrogen control system suitable for passive plants, which would also meet the regulations. The ALWR program intends to specify hydrogen control requirements consistent with NRC staff guidance. The passive ALWR vendors have indicated that their designs will comply with the applicable requirements document.

In its letter of August 17, 1992, ACRS indicated that it supported the staff's recommendation that hydrogen mitigation systems accommodate a 100-percent fuel-clad metal-water reaction. ACRS recommended that the staff perform an analysis similar to that conducted in support of the resolution of Generic Issue 106, "Piping and Use of Highly Combustible Gases in Vital Areas" on the impact of hydrogen combustion and possible detonation including stratification, before establishing an average hydrogen concentration limit.

The staff has performed numerous studies and conducted several experimental programs to better understand the behavior of hydrogen combustion and potential concentration gradients within the primary containment. While it is clear that additional analyses would add to the overall data base, the staff believes that a *sufficient base exists today to go forward with licensing criteria*. With respect to the insights gained from Generic Issue 106, the staff has extensively examined the potential for and consequences of detonations. In the last decade, the results of these efforts have been published in NUREG/CR-4905, 4961, 5275, and 5525. The current NRC research program is also examining the generic issue of local explosions and the initiation of detonations by jet flames.

<u>The staff recommends that the Commission approve the position that passive plants should be designed, as a minimum, to the same requirements applied to evolutionary designs, as approved by the Commission's SRM of June 26, 1990. Specifically, passive plant designs must:</u>

- <u>accommodate hydrogen generation equivalent to a 100-percent metal-water reaction of the fuel cladding;</u>
- <u>limit containment hydrogen concentration to no greater than 10 percent; and,</u>
- <u>provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents.</u>

H. Core Debris Coolability

In the unlikely event of a severe accident in which the core melts 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. Therefore, the staff concluded that plant designs should include features to enhance core debris coolability.

In SECY-90-016, the staff recommended that the Commission approve the general criteria that evolutionary ALWR designs:

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

In its SRM of June 26, 1990, the Commission approved the staff's position.

In addition, the staff indicated in SECY-90-016 that it was evaluating the level of protection afforded by covering the containment liner and other structural members with concrete. Debris coolability is an area in which there is active ongoing experimental research including relatively large scale testing jointly sponsored by EPRI and NRC. The results of tests were expected to demonstrate the early quenchability of core debris within the reactor cavity. However, these tests were indeterminate in proving quenchability and indicated the need to consider the potential for continued core-concrete interaction. Concrete can be used as a sacrificial barrier, for both the liner and structural components, to accommodate potential longer periods of core-concrete interaction. The staff now concludes that it may be necessary to protect these structural components with concrete.

The EPRI requirements document and the evolutionary ALWR designs provide a number of design features that are intended to mitigate the effects of a molten core. Among other features, the evolutionary designs proposed floor sizing criteria of 0.02 m^2/MWt, and provisions to flood the lower drywell or reactor cavity. The staff neither supports nor disputes the EPRI floor sizing criteria of 0.02 m^2/Mwt. Instead, the staff concludes that it is appropriate to review the specific vendor designs to determine how they address the general criteria discussed above (including protecting structural components with concrete) to provide an increased level of protection relative to core debris coolability. *The staff concludes that the "core-on-the-floor" accident will not be considered as a new design-basis accident.* However, the staff expects the vendors to consider the effects of core-concrete interactions on the production of non-condensible gases, the release of additional fission products from the core-concrete interaction, and additional heat and hydrogen generation in the new designs.

The criteria discussed above are intended to ensure that the ALWR vendors provide measures (to the extent practical) to mitigate severe accidents, while avoiding turning severe accidents into traditional design-basis accidents. Because the staff neither supports nor disputes particular floor sizing criteria, vendors should ensure that the containment can withstand the environmental conditions (pressure and temperature) and structural challenge caused by core-concrete interactions. For the range of severe accidents of concern, vendors should realistically estimate the amount of core-concrete interaction that will occur, and ensure that the containment will accommodate the resultant conditions for approximately 24 hours without loss of containment integrity. Where insufficient data exist to develop realistic estimates, vendors may propose alternatives (such as additional tests or the use of other methodologies) for determining the degree of core-concrete interaction. The ALWR vendors should also perform parametric studies to determine how sensitive the containment response is to variations in the amount of core-debris available to interact with the concrete.

The staff concludes that incorporation of mitigative measures (to the extent practical) and assurance of containment integrity for approximately 24 hours will provide defense-in-depth as well as an appropriate degree of robustness in the containment design.

In its letter of May 5, 1992, EPRI indicated that the requirements documents specify requirements to address debris coolability including cavity/lower drywell area to permit spreading, cavity/lower drywell flooding to quench debris, and protection of the containment boundary from debris attack. The requirements documents also specify that containment loads from dominant core damage sequences be evaluated. The passive ALWR vendors have indicated that their designs will comply with the applicable EPRI requirements document.

<u>The staff recommends that the Commission approve the position that both the evolutionary and passive LWR designs meet the following criteria:</u>

- <u>provide reactor cavity floor space to enhance debris spreading;</u>

- <u>provide a means to flood the reactor cavity to assist in the cooling process;</u>

- <u>protect the containment liner and other structural members with concrete, if necessary; and,</u>

- <u>ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed Service Level C for steel containments or Factored Load Category for concrete containments, for approximately 24 hours. Ensure that the containment capability has margin to accommodate uncertainties in the environmental conditions from core-concrete interactions.</u>

I. High-Pressure Core Melt Ejection

In SECY-90-016, the staff recommended that the Commission approve the position that evolutionary ALWR designs should include a depressurization system and cavity design features to contain ejected core debris in order to reduce the potential for containment failure as a result of direct containment heating (DCH). The staff is concerned that this event might result from the ejection of molten core debris under high-pressure from the reactor vessel. Such an ejection might result in wide dispersal of core debris, rapid oxidation, and extremely rapid addition of energy to the containment atmosphere.

In its SRM of June 26, 1990, the Commission approved the staff's position. The Commission also directed that the cavity design, as a mitigating feature, should not unduly interfere with operations, including refueling, maintenance, or surveillance activities. Examples of cavity design features that will decrease the amount of ejected core debris that reaches the upper containment include (1) ledges or walls that would deflect core debris and (2) an indirect path from the lower reactor cavity to the upper containment. The staff will review the LWR designs relative to the above criteria.

In its letter of May 5, 1992, EPRI indicated that the requirements document specifies an RCS depressurization system and cavity retention capability for both evolutionary and passive plants. EPRI further indicated that since the passive plant emergency core cooling system (ECCS) relies on RCS depressurization, redundancy and diversity have been specified for the depressurization system to ensure very high reliability.

In its letter of August 17, 1992, ACRS indicated that because direct containment heating is an extremely improbable event, two modes of coping with the possibility are not needed. ACRS stated that because of the possible safety benefits for other events, reliable depressurization is the preferred approach.

The staff agrees with the ACRS assertion that a reliable depressurization system is needed. However, the staff proposes to provide a design concept with a degree of consequence mitigation along with a certain amount of accident prevention. The depressurization system retains a degree of uncertainty. Such questions as the rate of depressurization, the timing for operator initiation of manual depressurization, and the cut-off pressure may never be totally resolved. As a result, the staff believes that a design can be developed to decrease the direct flight path to the upper containment at little or no added expense.

The plant designers have provided features to address this issue for evolutionary ALWR designs. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's guidance on this issue. The staff's preliminary review of the passive ALWRs has also identified the importance of RCS depressurization to the safe shutdown of the plant during transients or accidents. RCS depressurization is crucial to the operation of the passive safety features that limit the likelihood of core damage, as well as to reducing the potential for containment failure by direct containment heating from the ejection of core debris at high pressure. Therefore, the staff has determined that the passive ALWR designs should include a highly reliable depressurization system.

<u>The staff recommends that the Commission approve the general criteria that the evolutionary and passive LWR designs</u>

- <u>provide a reliable depressurization system; and,</u>
- <u>provide cavity design features to decrease the amount of ejected core debris that reaches the upper containment.</u>

J. Containment Performance

As discussed in SECY-90-016, the staff recommended that the Commission approve the position to evaluate evolutionary ALWRs using a conditional containment failure probability (CCFP) of 0.1, or a deterministic containment performance goal that offers comparable protection. The staff concluded that the following general criterion would be an appropriate substitute for a CCFP in evaluating evolutionary ALWR containment performance during a severe-accident challenge:

> 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.

The staff proposed this containment performance goal to ensure that the containment will perform its function in the face of most credible severe accident challenges.

In its SRM of June 26, 1990, the Commission approved the use of a 0.1 CCFP as a basis for establishing regulatory guidance for the evolutionary ALWRs. The Commission directed, however, that this objective should not be imposed as a requirement, and that the use of the CCFP should not discourage accident prevention. The Commission also directed the staff to review and submit to the Commission suitable alternative, deterministically established containment performance objectives providing comparable mitigation capability that may be considered by the applicants.

In this paper and in SECY-90-016, the staff has identified the major challenges to the containment (such as hydrogen burns or core debris interactions with water and containment structures) and the need to provide the means for mitigation of these challenges. Nonetheless, the containment performance goal acts as defense-in-depth to ensure that the design (including its mitigation features) would be adequate if called upon to mitigate a severe accident.

Although not explicitly identified in SECY-90-016, the staff will also evaluate the impact of interaction between molten fuel and coolant, and the resulting steam and hydrogen generation (and any dynamic forces due to ex-vessel fuel-coolant interactions outside the vessel) on the integrity of the containment, consistent with the containment performance goal. The evaluation of containment bypass sequences will be addressed on a vendor-specific basis during the staff's review of ALWR designs.

The intent of both the CCFP and the alternative deterministic performance criteria discussed above is to provide a final check as well as defense-in-depth. The philosophy behind the use of the proposed deterministic goal is that adequate time must be provided for fission product decay before allowing a release from the containment to the environment.

In its letter of August 17, 1992, ACRS stated that the staff has not developed an adequate technical position relating to requirements for containment performance in passive designs. ACRS also contends that the CCFP approach should be used in developing regulatory requirements and not merely passed on to the applicants. With regard to deterministic evaluation of containment performance, ACRS implied that the staff should identify the containment challenges, or how they are to be quantified, on a generic basis. ACRS endorsed the criteria recommended in its letter of May 17, 1992, "Proposed Criteria to Accommodate Severe Accidents in Containment Design."

During the evolutionary ALWR reviews, the staff conducted a thorough review to ensure that a probabilistic CCFP would not be used in a way that could detract from a balanced approach of severe accident prevention and consequence mitigation. The ABWR review is nearly complete and, at this time, the staff believes GE has met the CCFP goal of 0.1 for internal events. The ABB-CE System 80+ design is also expected to meet the CCFP goal of 0.1.

Through the review of the evolutionary ALWR designs, the staff recognized the limitations of a CCFP approach. Specifically, as accident prevention is increased by decreasing core damage frequency, the ability of the containment to withstand events of even lower probability becomes less clear. The limitations of a CCFP approach are also evident in the uncertainties prevalent within a probabilistic risk assessment (PRA). Because of these limitations, vendors have provided some deterministic analysis to complement the CCFP approach by addressing uncertainties in the severe accident phenomena and calculation of the CCFP. In particular, uncertainties surrounding the issue of debris coolability have led the staff to conclude that deterministic best-estimate analyses should complement the CCFP approach. In spite of its limitations, the staff believes that the CCFP approach ensures that evolutionary ALWRs maintain a balance between accident prevention and consequence mitigation.

In its letter of May 5, 1992, EPRI indicated that the requirements document specifies a set of deterministic containment performance requirements supported by PRA. These requirements address a set of severe accident containment challenges and specify that ASME Service Level C limits be met for risk-significant sequences (expected to be low-pressure core melt into an intact containment). EPRI indicated that the proposed staff criteria are generally consistent with the ALWR requirements, except in the challenges to be considered in demonstrating the 24-hour Service Level C goal. EPRI has provided a containment performance study for both evolutionary and passive plant designs that identifies 23 postulated containment challenges and failure modes along with specification of design features for their mitigation and requirements for deterministic analysis. The staff has reviewed the listing of identified challenges and believes that it represents a complete set.

In SECY-92-292, "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future ALWRs," dated August 24, 1992, the staff proposed that the Commission issue an Advance Notice of Proposed Rulemaking (ANPR) on Severe Accident Plant Performance Criteria for Future LWRs. Intended to be used for passive LWRs, the ANPR contains three distinct options that could be followed to establish severe accident containment performance requirements. The three options include a hardware-oriented rule, a phenomena-oriented rule, and a general design criteria (GDC)-oriented rule that was outlined in detail in an ACRS letter dated May 17, 1991.

In an SRM dated September 17, 1992, the Commission directed the staff to place the ANPR in the Federal Register for a 90-day comment period. The staff will review the containment performance goal and severe accident guidance following receipt of all public comment and after meeting with Westinghouse and GE representatives to discuss their passive LWR designs. The staff will also discuss these issues with ACRS.

The staff is currently evaluating the containment performance of the individual ALWR designs to ensure that all potential sequences that may be identified during the staff's review are adequately addressed. The staff will evaluate the criteria used by the vendor to determine the challenges to the containment.

In SECY-90-016, the staff indicated that 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. Because both containment performance goals are acceptable and given the limitations inherent within the CCFP approach, the staff believes the deterministic containment performance goal should be pursued for the passive ALWR designs.

Although the staff is still evaluating this issue and expects further insight from public comments on the ANPR, the staff concludes that it is appropriate to proceed with interim severe accident containment performance criteria. These interim criteria are intended to reflect the lessons learned during the review of the evolutionary designs. The staff recommends that the Commission approve the position to use the following deterministic containment performance goal in the evaluation of the passive ALWRs:

> The containment should maintain its role as a reliable, leak-tight barrier (for example, by ensuring that containments stresses do not exceed ASME Service Level C limits for metal containments, or Factored Load Category for concrete containments) approximately 24 hours following the onset of core damage under the more likely severe accident challenges and, following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

The staff will inform the Commission if it identifies additional policy issues as a result of (1) reviewing ALWR designs, or (2) evaluating comments on the ANPR concerning "Severe Accident Plant Performance Criteria for Future LWRs."

K. Dedicated Containment Vent Penetration

In SECY-90-016, the staff recommended that the Commission approve the use of an overpressure protection system that uses a dedicated containment vent for the ABWR. This system should be designed to avoid gross containment failure resulting from postulated slow rising overpressure scenarios that could result from postulated multiple safety system failures.

In its SRM of June 26, 1990, the Commission approved the staff's recommended use of the containment overpressure protection system for the ABWR, subject to a comprehensive regulatory review to consider the "downside" risks with the mitigation benefits of the system. In addition, the Commission directed the staff to ensure that the design provides full capability to maintain control over the venting process.

In its letter of May 5, 1992, EPRI indicated that containment venting was an optimization issue in the EPRI requirements documents. According to the ALWR Utility Steering Committee representatives, specific containment overpressure protection for evolutionary ALWR designs, either through the size and strength of containment or through installation of an overpressure protection relief, is considered an acceptable approach.

EPRI indicated that passive plant design features that address the containment overpressure challenge include highly reliable, redundant, and diverse passive safety-grade decay heat removal, automatic depressurization, and containment cooling. EPRI recommended that the NRC not require a containment vent in evolutionary PWR containments and passive ALWRs.

In its letter of August 17, 1992, ACRS indicated that the Commission should make a generic judgement about the acceptability of containment vents for light water reactors. ACRS contends that this should be part of establishing general criteria for containment designs, as proposed in their letter of May 17, 1992.

The staff considers the containment vent as one of many plant systems that can be used to mitigate the consequences of an accident. If acceptable analyses indicate that a vent would not be needed to meet the severe accident criteria, such as the containment performance goal, the staff would not propose to implement a vent requirement.

Because of the current stage of design development and review, the staff has insufficient information at this time to determine whether a containment vent is necessary for passive plant designs. The containment performance criteria proposed in Section I.J of this enclosure will serve as the basis for the staff's review of containment integrity and the need for a containment vent. Therefore, the staff recommends that the Commission approve the position that the need for a containment vent for the passive plant designs should be evaluated on a design-specific basis.

L. Equipment Survivability

In SECY-90-016, the staff recommended that the Commission approve the position that features provided only for severe-accident protection need not be subject to the environmental qualification requirements of 10 CFR 50.49; quality assurance requirements of 10 CFR Part 50, Appendix B; or redundancy/diversity requirements 10 CFR Part 50, Appendix A. The reason for this judgement is that the staff does not believe that severe core damage accidents should be treated in the same manner traditionally used for design-basis accidents (DBAs) because of significant differences in their likelihood of occurrence. However, SECY-90-016 further stated that mitigation features must be designed to provide 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 provided for DBAs is relied upon to cope with severe-accident situations, there should also be a high confidence that this equipment will survive severe-accident conditions for the period that it is needed to perform its intended function.

During the review of the credible severe-accident scenarios for ALWR designs, the staff will evaluate the ALWR vendors identification of the equipment needed to perform mitigative functions and the conditions under which the mitigative systems must operate. Equipment survivability expectations under severe-accident conditions should consider the circumstances of applicable initiating events (such as station blackout or earthquakes) and the environment (including pressure, temperature, and 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 its SRM of June 26, 1990, the Commission approved the staff's position. In its letter of May 6, 1991, the staff clarified its position that these criteria would be applied to features provided only for severe accident mitigation.

The EPRI requirements document and the evolutionary ALWR designers have indicated that their submittals are consistent with these criteria. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's guidance on this issue, but does not expect any related policy matters to result from its review. The passive ALWR vendors have indicated that their designs will comply with the applicable EPRI requirements document. In its letter of August 17, 1992, ACRS agreed with the staff position discussed above.

Although the staff is still evaluating this issue for the passive plant designs, it has not identified any unique features of the designs that would preclude the staff's conclusion that these designs should also be evaluated against the criteria established for evolutionary plants. <u>Therefore, the staff recommends that the Commission approve the position that passive plant design features provided only for severe-accident mitigation need not be subject to the environmental qualification requirements of 10 CFR Section 50.49; quality assurance requirements of 10 CFR Part 50, Appendix B; and redundancy/diversity of requirements 10 CFR Part 50, Appendix A.</u> As discussed in SECY-90-016, the staff concludes that guidance such as that found in Appendices A and B of RG 1.155, "Station Blackout," is appropriate for equipment used to mitigate the consequences of severe accidents.

M. Elimination of Operating-Basis Earthquake

In SECY-90-016, the staff discussed its proposal to decouple the operating-basis earthquake (OBE) from the safe-shutdown earthquake (SSE). The regulations in Appendix A to 10 CFR Part 100 establish the OBE at one-half the intensity of the SSE. The staff stated that the OBE should not control the design of safety systems and was evaluating possible changes to the regulations that would reduce the magnitude of the OBE relative to the SSE. The staff recommended that the Commission approve the review approach to consider requests to decouple the OBE from the SSE on a design-specific basis for evolutionary designs. In its SRM of June 26, 1990, the Commission approved the staff's recommendation.

In a plant optimization subject, EPRI requested that the staff evaluate the total elimination of the OBE from the design of systems, structures, and components (SSC) in nuclear power plants. In its letter of April 26, 1990, the ACRS also recommended this approach. In evaluating the decoupling of the OBE from the SSE, the NRC staff is also evaluating the possibility of redefining the OBE in order to satisfy its function without an explicit response analysis. This change would diminish the role of the OBE in design by establishing a level which, if exceeded, would require that the plant be shut down for inspection activities.

EPRI's position on seismic design is that it is unnecessary to perform two complete sets of seismic analyses -- one for the OBE and one for the SSE. The NRC staff agrees, in principle, with this position, but finds that extant design practices for piping and structures do not result in designs that are significantly controlled by the OBE. As stated in SECY-90-016, certain interim measures (such as allowing higher damping values for piping analyses) have already been implemented to mitigate the situation of having the OBE significantly control the design.

The elimination of the OBE response analysis would require performance of all current OBE design-related checks for a fraction of the SSE. The staff is working with the industry to develop various design alternatives for the ABWR to supplement the codes and standards when design-related checks are based on the OBE. For example, in piping design, the ASME Boiler and Pressure Vessel Code currently establishes rules for evaluating earthquake cycles on fatigue and relative seismic anchor motion effects that are based on the OBE. The staff and industry are supplementing the Code with design rules that would account for fatigue and seismic anchor motion effects based on the SSE. In addition, the NRC guidelines for postulating the number and location of pipe ruptures are also derived from the OBE. When the OBE is eliminated from design, these loading calculations may need to be performed using the SSE after establishing appropriate new allowable limits.

The staff's proposed amendment of Appendix A to 10 CFR Part 100 would allow the option to eliminate the OBE from design certification when the OBE is established at less than or equal to one-third the SSE. In this manner, the OBE serves as an "inspection level earthquake" below which the effect on the health and safety of the public would be insignificant and above which the licensee would be required to shut down the plant and inspect for damage.

The staff assessed the safety margins of several aspects of nuclear plant design when the OBE is eliminated from consideration. The industry and staff recognize that earlier seismic criteria caused the OBE to control certain aspects of the plant design (such as the piping systems). The industry and the staff view the "controlling" nature of the OBE design as an additional margin above the safety margins established by the design bases. Therefore, eliminating the OBE would not result in a significant decrease in the overall plant safety margin. The staff is currently performing a detailed evaluation of the extent to which the OBE controls the design and the effect on the design of SSC when the OBE is eliminated from design consideration.

The overall design of reactor site structures is generally conservative, and the structural responses for all combinations of loads (including those from earthquakes) are kept at or below the material yield stresses to preclude plastic deformation.

The staff has examined the structural load combinations and the corresponding acceptance criteria. On the basis of analyses, tests, and engineering judgment, the staff has determined that the structural design produced using SSE load combinations envelop the load combinations produced using the OBE. These conclusions are consistent with the staff's licensing experience accumulated during its review of many seismic calculations for individual plants and test data from NRC-sponsored research.

For analyses of safety-related structures, the effects of the relative displacements between adjacent structures need to be considered for earthquakes. With the elimination of the OBE, these effects should be considered for the SSE and would also be needed for input into the piping design as discussed later in this issue. In RGs 1.27 and 1.143, the staff recommends that the designs of the ultimate heat sink and radioactive waste structures, respectively, be evaluated only for OBE loading using the appropriate load combinations and limits for the OBE. When the OBE is eliminated from the ABWR, the staff found that GE proposed to design these buildings and structures to the SSE loadings. The staff concluded that using SSE loadings and the appropriate load combinations and limits provides a bounding design comparable to that provided in the regulatory guides. The staff will also review alternative methods that might be proposed by other vendors on a case-by-case basis.

A designer of piping systems considers the effects of primary and secondary stresses and evaluates fatigue caused by repeated cycles of loading. Primary stresses are induced by the inertial effects of vibratory motion. The relative motion of anchor points induces secondary stresses. The repeating seismic stress cycles induce cyclic effects (fatigue).

After reviewing these aspects, the staff concludes that, for primary stresses, if the OBE is established at one-third the SSE, the SSE load combinations control the piping design when the earthquake contribution dominates the load combination. Therefore, the staff concludes that eliminating the OBE piping stress load combinations for primary stresses in piping systems will not significantly reduce the existing safety margins.

Eliminating the OBE will, however, directly affect the current methods used to evaluate the adequacy of cyclic and secondary stress effects in the piping design. Eliminating the OBE from the load combination could cause uncertainty in evaluating the cyclic (fatigue) effects of earthquake-induced motions in piping systems and the relative motion effects of piping anchored to equipment and structures at various elevations because both of these effects are currently evaluated only for OBE loadings.

Accordingly, to account for earthquake cycles in the fatigue analysis of piping systems, the staff proposes to develop guidelines for selecting a number of SSE cycles at a fraction of the peak amplitude of the SSE. These guidelines will provide a level of fatigue design for the piping equivalent to that currently provided in the standard review plan (SRP)(NUREG-0800). Currently, the staff's guidelines in SRP Section 3.9.2 recommend an equivalent of 50 OBE peak cycles for fatigue evaluation. The staff will develop new guidelines after conducting regulatory research and will incorporate the guidelines into an SRP revision or into a regulatory guide, as necessary.

To account for earthquake cycles in the fatigue analyses of piping systems performed until the new guidance is issued, the staff proposes using two SSE events with 10 maximum stress cycles per event (20 full cycles of the maximum SSE stress range). This is equivalent to the cyclic load basis of one SSE and five OBE events, as currently recommended in SRP Section 3.9.2, when accounting for differences in the structural damping between the OBE and SSE and for a 60-year (instead of a 40-year) plant life. Alternatively, the staff proposes that number of fractional vibratory cycles equivalent to that of 20 full SSE vibratory cycles may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with Appendix D of IEEE Standard 344-1987.

The ASME Boiler and Pressure Vessel Code (ASME Code), Section III, Paragraph NC/ND-3655 specifies that seismic anchor displacement effects need not be considered for Service Level D. However, the ASME Code requires that seismic anchor motion stresses be considered for Service Level B, for which the OBE has traditionally been the designated seismic loading. If the OBE were eliminated from the piping design, the ASME Code, Section III evaluation would have no requirement for considering the effects of seismic anchor motion. The staff proposes that the effects of anchor displacements in the piping caused by an SSE be considered with the Service Level D limit. The staff's recommendation will correct this anomaly and will require an evaluation of seismic anchor motion effects for the SSE together with the effects of normal conditions as required by 10 CFR Part 50, Appendix A, GDC 2. Their effects would be evaluated to a Service Level D limit for which the SSE has traditionally been the designated seismic loading.

The staff proposes that existing staff guidelines ensuring the functionality of safety-related components and supports under SSE loading conditions be maintained. For example, when safety-related equipment is qualified by analysis only, the stress limit should remain in the linear-elastic range for SSE loading. Similarly, the function of the supported system must be taken into account. As specified in RG 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Components Supports," the ASME Service Level B limits of Subsection NF (or other justifiable limits approved by the staff) should be used to ensure that systems which normally prevent or mitigate consequences of SSE-related events will operate adequately regardless of plant condition.

Pipe rupture is a rare event that can be caused by errors in design, construction, or operation; unanticipated loads; or unanticipated corrosive environments. The staff notes that piping failures generally occur at high stress and fatigue locations, such as at the ends of a piping system where it connects to component nozzles.

Recent dynamic pipe tests conducted by EPRI and the NRC demonstrated that butt-welded piping can withstand seismic inertial loadings higher than an SSE without rupturing. Thus, the staff concludes that the likelihood of a double-ended pipe rupture caused by an OBE-level earthquake in a piping system designed to an SSE is remote. Operating experience has shown that pipe failures (splits, through-wall cracks, and double-ended pipe ruptures) are more likely to occur under conditions caused by normal operation. These conditions include erosion, corrosion, thermal constraint, fatigue, and operational transients.

The staff recommends that the Commission approve the approach to eliminate the OBE from the design of systems, structures, and components. When the OBE is eliminated from the design, no replacement earthquake loading should be used to establish the postulated pipe rupture and leakage crack locations. The staff recommends that the criteria for postulating pipe ruptures and leakage cracks in high- and moderate-energy piping systems be based on factors attributed to normal and operational transients alone. However, for establishing pipe breaks and leakage cracks due to fatigue effects, calculation of the cumulative usage factor should continue to include seismic cyclic effects.

Further reduction in the number of postulated pipe rupture locations can be considered when compensatory measures are established to minimize the potential for pipe ruptures during normal operating and transient conditions (such as control of erosion/corrosion or use of upgraded piping materials). The guidelines for environmental qualification and compartment pressurization are currently based on the mechanistic break locations. Therefore, the staff proposes that the mechanistic pipe break and high-energy leakage crack locations determined by the piping high stress (without the OBE) and fatigue locations may be used for equipment environmental qualification and compartment pressurization purposes.

Eliminating the OBE from explicit design consideration affects several aspects of the seismic qualification of safety-related mechanical and electrical equipment. When the equipment qualification is performed by analysis, the acceptance criteria are derived from the ASME Code. The effect of eliminating the OBE from equipment qualification by analysis should be negligible. It is well known that mechanical equipment (such as pumps and valves) is, in general, seismically rugged when adequately anchored. It is also known that operability limits for mechanical equipment are generally established through maximum permissible moments and forces or tolerance limits based on available clearances that are controlled by the SSE (rather than the OBE). Therefore, for mechanical equipment, elimination of OBE from qualification by analysis should not reduce any margin. Also, some electrical equipment may be qualified by an analysis requiring demonstration that five OBE events followed by one SSE event do not cause a failure of the equipment to perform its safety function. With the elimination of OBE, analysis checks for fatigue effects can be performed at a fraction of the SSE (such as 50 cycles at one-half of the SSE peak amplitude, or 150 cycles at one-third of the SSE peak amplitude).

When equipment qualification for seismic loadings is performed by analysis, testing, or a combination of both, the staff recommends the use of IEEE Standard 344-1987, as endorsed in RG 1.100, Revision 2. For such analysis, the selection of the service limit level for different loading combinations will ensure the functionality of the equipment during and following a SSE. For testing, IEEE Standard 344-1987 details requirements for performing seismic qualification using five OBE events followed by an SSE event. Where complex mathematical models are based solely on calculated structural parameters, verification testing should be performed.

With the elimination of the OBE, two alternatives exist that will essentially maintain the requirements provided in IEEE Standard 344-1987 to qualify equipment with the equivalent of five OBE events followed by one SSE event (with 10 maximum stress cycles per event). Of these alternatives, the staff concludes that equipment should be qualified with five one-half SSE events followed by one full SSE event. Alternatively, a number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance with Appendix D of IEEE Standard 344-1987 when followed by one full SSE. The staff will conduct research to verify the number of SSE cycles and the fraction of their peak amplitude for which the equipment is to be tested. The staff will also review the results of research conducted by the industry standards group, as appropriate.

Lastly, the design of ALWRs using a single-earthquake (that is, SSE) design is predicated on the adequacy of pre-earthquake planning and post-earthquake damage inspections that are to be implemented by the combined operating license (COL) applicant. The staff proposes that the COL applicant submit to the NRC staff, as a part of its application, the procedures it intends to use for pre-earthquake planning and post-earthquake inspections. The staff is currently developing a regulatory guide for pre-earthquake planning and post-earthquake operator actions.

In its letters dated May 5, 1992, and August 21, 1992, EPRI representatives commented on the staff's positions concerning eliminating the OBE from design as discussed in the two draft Commission papers. EPRI agrees with the staff's recommendation to eliminate the OBE from design. The EPRI requirements documents will require a seismic margins assessment, which demonstrates a margin for an earthquake substantially larger than the SSE. However, EPRI does not fully agree with the number of earthquake cycles to be used in fatigue evaluation and equipment qualification. The EPRI recommended that the NRC guidelines be similar to those contained in paragraph N-1214 of ASME Code, Section III, Appendix N, which provides that no more than two OBE events with a total of 20 full stress cycles (that is, 20 cycles at one-half SSE) be used.

Similarly, in its letter dated September 17, 1992, Westinghouse Electric Corporation representatives commented on the staff's position regarding the elimination of the OBE and noted that the staff's proposed position was overly conservative in the number of earthquake cycles to be considered. Westinghouse proposed to adopt the guideline of 20 cycles at one-half of the SSE response, as specified by the EPRI URD for passive plants.

The staff's evaluation indicates, however, that the EPRI and Westinghouse recommendations provide less stringent design requirements than those currently prescribed by the staff guidelines of SRP Section 3.9.2. Additionally, the current staff guidelines were used for the review of nuclear plants with a 40-year life or less.

The staff positions presented above have been revised since the draft Commission papers to account for both a 60-year plant life and the differences in the structural damping used for the OBE and SSE. (Component damping was not specified because EPRI pointed out that is was the same for the OBE and SSE in ASME Code Case N-411). The staff also found that the overall contribution of earthquake cycles to fatigue is small. To design for 20 full SSE cycles (or the equivalent) would not significantly penalize the design and would provide a bounding design for the expected number of earthquakes of a lesser magnitude than the SSE and their aftershocks for a 60-year plant life.

In its letter of September 16, 1992, ACRS stated that it believes the staff took an appropriate approach in its interim position (which has been incorporated into the final staff position above).

In its final position, the staff supplemented its preliminary positions from the draft Commission papers and provided a more complete package identifying the actions necessary for the design of SSC when the OBE is eliminated. As discussed above, the staff clarified that guidelines should be maintained to ensure the functionality of components, equipment, and their supports. In addition, the staff clarified how certain design requirements are to be considered for buildings and structures that are currently designed for the OBE earthquake, but not the SSE. Also, the staff addressed how pre-earthquake planning and post-earthquake operator actions are to be considered. The staff's proposed guidelines and their bases are discussed in the final positions above.

The staff has evaluated the effect on safety of eliminating the OBE from the design load combinations for selected SSC and has developed proposed criteria for an analysis using only the SSE. <u>The staff, therefore, requests that the Commission approve the proposed positions discussed above.</u> The staff will keep the Commission informed as the review progresses and will note in case-specific safety evaluations instances in which the applicant proposes to use criteria different than those described above for an SSE-only analysis.

N. Inservice Testing of Pumps and Valves

In SECY-90-016, the staff recommended that the Commission approve the position that 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 that cannot be detected through the use of advanced, non-intrusive techniques.

The staff concluded that these requirements are necessary to provide an adequate assurance of operability.

In its SRM of June 26, 1990, the Commission approved the staff's position as supplemented in the staff's response to ACRS comments, dated April 27, 1990. In that response, the staff agreed with the ACRS recommendations to emphasize the requirements of Generic Letter (GL) 89-10 with regard to evolutionary plants, to resolve check valve testing and surveillance issues, and to indicate how these requirements are to be applied to evolutionary plants. The staff also agreed that the requirements should permit consideration of proposed alternative ways of meeting inservice and surveillance requirements. The Commission further noted that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

After reviewing the proposed staff requirement in SECY-90-016 which stated that designs should incorporate provisions to test motor-operated valves under design-basis differential pressure, the staff has clarified its position. The staff recommends that full flow testing be conducted at maximum design flow with analysis to extrapolate to design pressure if it is not practicable to conduct the inservice pump testing at design flow and pressure. The staff also recommends that for valves, a qualification test (under design-basis differential pressure) be conducted prior to installation and inservice valve tests be conducted under the maximum practicable differential pressure and flow when it is not practicable to achieve design-basis differential pressure during an inservice test.

In its letter of May 5, 1992, EPRI stated that the ALWR program agrees with the above staff positions for the passive and evolutionary plants. In its letter of August 17, 1992, ACRS stated that they support the staff's recommendation that the above design, testing, and inspection provisions should be imposed on all safety-related pumps and valves for passive ALWRs.

EPRI and the evolutionary ALWR designers have indicated that their submittals are consistent with these criteria. The staff is in the process of evaluating their submittals to ensure acceptable implementation of the Commission's guidance on this issue. The passive ALWR vendors have indicated that their designs will comply with the applicable EPRI requirements document.

Therefore, the staff recommends that the Commission approve the position that these requirements should also be imposed on passive ALWRs. The staff concludes that additional inservice testing requirements may be necessary for certain pumps and valves in passive plant designs. This necessity arises because the passive safety systems rely heavily on the proper operation of this equipment (such as check valves or depressurization valves) to mitigate the effects of accidents and to shut down the reactor. The staff will discuss its proposed resolution of this issue in a separate Commission paper addressing the regulatory treatment of nonsafety systems in passive plant designs.

II. Other Evolutionary and Passive Design Issues

A. Industry Codes and Standards

In SECY-91-273, "Review of Vendors' Test Program to Support the Design Certification of Passive Light-Water Reactors," dated August 27, 1991, the staff raised the concern that a number of design codes and industry standards dealing with new plant construction have recently been developed or modified, and that the NRC has not yet determined their acceptability.

The staff recommends that the Commission approve the position, consistent with past practice, that it will review both evolutionary and passive plant design applications using the newest codes and standards that have been endorsed by the NRC. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis. In its letter of May 13, 1992, ACRS agreed with the staff's position. Similarly, in its letter of May 5, 1992, EPRI stated that the staff's position is consistent with the EPRI requirements documents.

B. Electrical Distribution

In SECY-91-078, "Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light-Water Reactor (LWR) Certification Issues," dated March 25, 1991, the staff recommended that the Commission approve its position that an evolutionary plant design should include the following elements:

- an alternative power source to the non-safety loads unless the design can demonstrate that the design margins will result in transients for a loss of non-safety power event that are no more severe than those associated with the turbine-trip-only event in current existing plant designs; and,

- at least one offsite circuit to each redundant safety division supplied directly from one of the offsite power sources with no intervening non-safety buses in such a manner that the offsite source can power the safety buses upon a failure of any non-safety bus.

In its SRM of August 15, 1991, the Commission approved the staff's positions.

In its letter of May 5, 1992, EPRI indicated that this issue is not applicable to passive designs. However, the staff has not yet determined the applicability of this issue to the passive designs. This issue will be addressed in a separate Commission paper which will discuss the regulatory treatment of active nonsafety systems in passive plant designs.

C. Seismic Hazard Curves and Design Parameters

To assess the seismic risk associated with an ALWR design, EPRI has proposed the use of generic bounding seismic hazard curves for sites in the central and eastern United States. EPRI proposes that these curves be used in the seismic PRA. Current regulations do not require that a seismic PRA be performed to determine if a site is acceptable, and the staff does not intend to require such an assessment.

To assess the EPRI ALWR seismic hazard bounding curve for rock sites, the staff compared the EPRI curve to results derived by Lawrence Livermore National Laboratories (LLNL). For this comparison, the staff used the historical earthquake method discussed in NUREG/CR-4885, "Seismic Hazard Characterization of the eastern United States: Comparative Evaluation of the LLNL and EPRI Studies," 1987. The staff also compared the EPRI bounding curve to hazard curves generated by EPRI using the historical method for the Seabrook site (see letter dated October 17, 1991). The historical hazard curves below 0.1g reflect the past few hundred years of historical earthquake data. The historical hazard curves at higher accelerations are estimates based on the historical earthquake data. Both the LLNL and EPRI hazard curves, which were derived using the historical method, exceed the EPRI bounding curve at accelerations below about 0.1g. Because the EPRI bounding curve is exceeded at low peak accelerations by the results based on historical earthquake data, the staff also questions the adequacy of the EPRI bounding curve at higher peak accelerations.

Hazard curves generated for the Seabrook Station Probabilistic Safety Assessment (1983) by the licensee also exceed the EPRI bounding hazard curve. The Seabrook SSE has a peak acceleration of 0.25g, whereas a higher SSE of 0.3g is proposed for ALWR sites. Based on the deterministic design basis of 0.3g, the EPRI-proposed criteria can be assumed to be suitable for the Seabrook site. However, based on the probabilistic assessment, the EPRI bounding hazard curve would underestimate the core damage frequency. Thus, the EPRI bounding hazard curve is non-conservative when compared to a licensee submittal.

Similarly, the LLNL hazard curves used in the staff's reviews of seismic hazard are generally higher than the EPRI results for the same sites. Some LLNL hazard curves for sites in the Eastern United States (discussed in NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains," 1989) exceed the EPRI bounding hazard curve.

During the staff's review of the ABWR, PRA results using both LLNL and EPRI hazard estimates were compared with results using the ABWR bounding seismic hazard curve. The ABWR bounding hazard curve was exceeded by the LLNL mean hazard curves for the Pilgrim, Seabrook, and Watts Bar sites. These three sites in the eastern United States were selected because of their relatively high seismic hazard. The staff used both LLNL and EPRI seismic hazard estimates to quantify core damage frequency. The PRA using the LLNL hazard curves predicted much higher core damage frequencies than the PRA using the EPRI hazard curve. However, the ABWR design was determined to be capable of resisting earthquakes significantly larger than an SSE of 0.3g.

The evolutionary and passive ALWR designers have indicated that their applications will be consistent with the EPRI criteria. However, based on review of historical seismicity and the LLNL hazard estimates, the staff concludes that the EPRI seismic hazard bounding curve is not sufficiently conservative. The staff is evaluating the seismicity and ground motion inputs used in the LLNL and EPRI studies to determine if the uncertainties in the curves can be reduced.

To judge the seismic capability of the GE and ABB-CE designs for sites in the continental U.S., the staff used a deterministic process. On that basis, the staff concludes that, with few exceptions, most areas of the U.S. would be candidate sites for these designs. As part of the COL process, the applicant will have to demonstrate that the site-specific seismic parameters are within the bounding site parameters for the certified design.

In its letter of May 5, 1992, EPRI stated that they now specify a Seismic Margins Assessment (SMA) methodology, which plant designers can use to assess the capability of advanced plants to shut down following a seismic event greater than an SSE. In addition, the PRA methodology may be used in evaluating a balanced seismic capability of standard designs, but calculation of risk as part of an overall core damage frequency determination is not required. The staff has reviewed this SMA methodology as part of its review of the EPRI requirements documents and should resolve any issues associated with this methodology through resolution of open items in the safety evaluation reports.

The discussion on seismic hazard curves provided in this section is for information only. If a policy question is identified as a result of its review, the staff will inform the Commission of the issue at the earliest opportunity.

D. Leak-Before-Break

GDC 4 states, in part, that "dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

Under the broad scope revision to GDC 4 (52 FR 41288, dated October 27, 1987), the NRC allows the use of advanced technology to exclude from structural design consideration the dynamic effects of pipe ruptures in nuclear power plants. However, it must first be demonstrated that the probability of pipe rupture is extremely low under conditions consistent with the design bases for the piping. Demonstration of low probability pipe rupture requires a deterministic fracture mechanics analysis that evaluates the stability of postulated small, through-wall flaws in piping and the ability to detect leakage through the flaws long before the flaws could grow to unstable sizes. The concept underlying such analyses is referred to as "leak-before-break" (LBB).

To date, the NRC staff has approved the LBB approach for currently operating and near-term operating licensed nuclear power plants based on a case-by-case review of plant-specific analyses. The NRC staff has approved the use of LBB for PWR primary coolant loop piping in all but five units in the United States. In addition, the use of LBB for pressurizer surge, accumulator, and residual heat removal piping have been approved for 11 units. In all cases, the LBB approvals have been granted for piping inside primary containment and for piping of at least 6 in. nominal diameter. The piping includes both austenitic and carbon steel material. However, all of the LBB-approved carbon steel piping has been clad with stainless steel material. To date, no BWRs have requested LBB approval.

EPRI and GE have proposed to adopt the LBB approach for ALWRs when certain details of the piping design, material properties, and stress conditions are known. As discussed in SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)," dated January 19, 1989, the staff will evaluate the acceptability of the use of LBB considerations in the ALWR designs when it can be justified. The staff has evaluated the EPRI and GE proposal for LBB application to ALWRs, as discussed below.

LBB Acceptance Criteria

The staff concludes that the analyses referred to in GDC 4 should be based on specific data, such as piping geometry, materials, and piping loads. The staff must review the LBB analyses for specific piping designs before the applicant can exclude the dynamic effects from the design basis. For ALWRs seeking design certification under 10 CFR Part 52, the analyses may be allowed to incorporate preliminary stress analysis results, provided that both upper- and lower-bounding limits are determined. Such limits would ensure that adequate margins are available for leakage, loads, and flaw sizes.

A leakage margin of 10 is required to ensure that leakage from the postulated flaw size is detected when the pipe is subjected to normal operational loads. A load margin of 1.4 is required to ensure that leakage-size flaws are stable at normal plus accident loads (such as SSE and safety-relief valve discharge loads). A factor of 2 between the leakage-size flaw and the critical-size flaw is required to ensure an adequate stability margin for the leakage-size flaw.

In addition, for ALWRs that seek NRC approval of LBB during the design certification phase, certain information will be required for LBB analyses to establish through-wall flaw sizes and flaw stability. For through-wall flaw sizes, a lower-bound, normal-operational stress limit must be established for dead weight, pressure, and thermal loadings. The mean or best-estimate stress-strain curve should be used. For flaw stability, an upper-bound stress limit should be established for normal loadings plus safe SSE and suppression pool hydrodynamic loadings. A lower-bound stress-strain curve for base metal should be used, regardless of whether the weld or base metal is limiting. In addition, a lower-bound toughness for the weld metal should be used.

A deterministic fracture mechanics evaluation accounting for material toughness is also required. An applicant may propose any fracture mechanics evaluation method for NRC staff review. However, the applicant will have to demonstrate the accuracy of the method by comparing it with other acceptable methods or with experimental data.

Using this approach, an initial set of bounding values and a preliminary LBB analysis can be established for ALWRs during the design certification phase. These bounding values and preliminary analyses can be verified when as-built and as-procured information becomes available during the COL phase. Before fuel-loading, the preliminary LBB analyses should be completely verified and based on actual material properties and final, as-built piping analysis as part of the inspections, tests, analyses, and acceptance criteria (ITAAC) associated with 10 CFR Part 52.

LBB Limitations

Because of the dependency of the LBB analyses to accurately predict the flaw stability, the NRC has established certain limitations for excluding from the LBB approach piping that is likely to be susceptible to failure from various degradation mechanisms during service. A significant portion of the LBB review involves evaluating the susceptibility of the candidate piping in various degradation mechanisms to demonstrate that the candidate piping is not susceptible to failure from these degradation mechanisms. The NRC staff reviews the operating history and measures to prevent or mitigate these mechanisms.

The LBB approach cannot be applied to piping that can fail in service from such effects as water hammer, creep, erosion, corrosion, fatigue, thermal stratification, and environmental conditions. The rationale is that these degradation mechanisms challenge the assumptions in the LBB acceptance criteria. For example, (1) water hammer may introduce excessive dynamic loads which are not accounted for in the LBB analyses, and (2) corrosion and fatigue may introduce flaws of a geometry that may not be bounded by the postulated through-wall flaw in the LBB analyses. Adhering to the "defense-in-depth" principle, piping susceptible to failure from these potential degradation mechanisms is excluded from LBB applications.

Alternatively, features to mitigate the possibility of certain degradation mechanisms may be proposed to ensure that LBB assumptions are not invalidated. For example, LBB might be considered for carbon steel piping for which the effects of erosion and corrosion have been eliminated through the use of high chromium steels with proven resistance to erosion and corrosion or through the use of carbon steel piping that is clad on the fluid-contacting surface with materials resistant to erosion and corrosion.

A detailed discussion of the limitations and acceptance criteria used for LBB by the NRC staff is provided in NUREG-1061, Volume 3, "Evaluation of Potential for Pipe Breaks: Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," dated November 1984.

Design Basis with LBB

The broad scope rule introduced an acknowledged inconsistency into the design basis by excluding the dynamic effects of postulated pipe ruptures while retaining non-mechanistic pipe rupture for the containment, ECCS, and environmental qualification (EQ) of safety-related electrical and mechanical equipment. The NRC staff subsequently clarified its intended treatment of the containment, ECCS, and EQ in the context of LBB application in a request for public comments on this issue that was published on April 6, 1988 (53 FR 11311).

Effects resulting from postulated pipe ruptures can generally be divided into local dynamic effects and global effects. Local dynamic effects are uniquely associated with a particular pipe rupture. These specific effects are not caused by a failure of any other source or even a postulated pipe rupture at a different location. Examples of local dynamic effects are pipe whip, jet impingement, missiles, local pressurization, pipe-break reaction forces, and decompression waves in the intact portions of that piping or communicating piping. Global effects of a pipe rupture need not be associated with a particular pipe rupture. Similar effects can be caused by failures of such sources as pump seals, leaking valve packings, flanged connections, bellows, manways, rupture disks, and ruptures of other piping. Examples of global effects are gross pressurization, temperatures, humidity, flooding, loss of fluid inventory, radiation, and chemical condition. For the ABWR, global effects also include suppression pool hydrodynamic loads (such as safety-relief valve discharges, pool-swell/fallback, condensation oscillation, and chugging loads).

The suppression pool hydrodynamic loads caused by a main steam or feedwater pipe rupture might be excluded for the design of piping, equipment, and internal containment structures (other than those serving a containment function). Nonetheless, the possibility of such dynamic effects being caused by a reactor internal pump ejection, failures of flanged connections, and blowdowns from ruptured disks or squib-actuated valves have not been addressed at this time. The option does exist to establish a postulated pipe break of a high-energy line smaller than the main steam or feedwater line break to envelop the possible global dynamic effects described above. The designer would be required to submit this approach for NRC staff review and approval before use. Until then, the use of a postulated pipe rupture of a main steam or feedwater line should be assumed for suppression pool hydrodynamic loads.

The application of LBB technology eliminates the local dynamic effects of postulated pipe ruptures from the design basis. Because the global effects from the postulated pipe rupture provide a convenient and conservative design envelope, the NRC staff will continue to require the consideration of global effects for various aspects of the plant design, such as environmental qualification of equipment, design of containments, and design of subcompartment enclosures.

Recommendations

The revised GDC 4 of Appendix A to 10 CFR Part 50 permits elimination of the dynamic effects of postulated high-energy pipe ruptures from the design basis of ALWRs using advanced fracture mechanics analyses. The limitations and acceptance criteria for LBB applications in ALWRs are the same as those established for currently operating nuclear power plants. <u>Therefore, the staff recommends that the Commission approve the application of the LBB approach to both evolutionary and passive ALWRs seeking design certification under 10 CFR Part 52. This approval should be limited to instances in which appropriate bounding limits are established using preliminary analysis results during the design certification phase and verified during the COL phase by performing the appropriate ITAAC discussed herein.</u> However, the specific details will need to be developed as the process is implemented during the first trial application.

In its letter dated May 13, 1992, ACRS agreed with the staff's recommendation to extend the application of the LBB approach to both evolutionary and passive ALWR plants. Similarly, in its letter dated May 5, 1992, EPRI agreed to extend the LBB approach to both evolutionary and passive plants and has since revised its URD to be consistent with the final staff's position stated above. The staff has not revised this position on LBB since the draft Commission papers.

E. Classification of Main Steamlines in Boiling Water Reactors (BWR)

The main steamlines in BWR plants contain dual quick-closing main steam isolation valves (MSIVs). These valves isolate the reactor system in the event of a break in a steamline outside of the primary containment, a design basis LOCA, or other events requiring containment isolation. Although the MSIVs are designed to provide a leak-tight barrier, it is recognized that some leakage through the valves will occur.

The current procedure for determining the acceptability of MSIV leakage involves calculating the dose in accordance with 10 CFR Part 100. This calculation is based on a conservative assumption that the leakage allowed by the technical specification (normally 11.5 SCFH per valve) is released directly into the environment. No credit is currently taken for the pressure integrity of the main steam piping and condenser.

Because of recurring problems with excessive leakage of MSIVs, the staff developed guidance in RG 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants." This RG recommends the installation of a supplemental leakage control system (LCS) to ensure that the isolation function of the MSIVs complies with the specified limits. However, operating experience has shown that the LCS has required substantial maintenance and resulted in substantial worker radiation exposure. Additionally, the NRC has generic concerns with the effectiveness of the LCS to perform its intended function under conditions of high MSIV leakage (Generic Issue C-8, Main Steam Line Valve Leakage Control Systems).

These concerns led EPRI to propose an alternative approach to ensure that doses associated with MSIV leakage would be acceptably low. EPRI identified this issue as a plant optimization subject. The resolution proposed by EPRI would allow higher leakage limits through the MSIVs, eliminate the safety-related leakage control system, and use an alternative MSIV leakage treatment method that takes credit for the large volume and surface area in the main steam piping and condenser hotwell to plate-out the fission products following core damage. In this way, the main steam piping, main steam drain and bypass lines, and condenser are used to mitigate the consequences of an accident.

Appendix A to 10 CFR Part 100 requires that SSCs necessary to ensure the capability to mitigate the consequences of accidents remain functional during and after a SSE. These components are classified as safety-related and seismic Category I. In addition, Appendix B to 10 CFR Part 50 establishes QA requirements for safety-related, seismic Category I SSCs.

Section 3.2.2 of the SRP recommends that the main steamline from the outermost isolation valve up to, but not including, the turbine stop valve including branch lines up to the first valve, be classified as Quality Group B (Safety Class 2). RG 1.29 designates such piping as seismic Category I.

The staff concludes that the main steam piping from the outermost isolation valve up to the seismic interface restraint should conform to Appendix A to SRP Section 3.2.2 and RG 1.29, as should branch lines 2½ inches in diameter and greater up to the first closed valve. The main steamline from the seismic interface restraint up to, but not including, the turbine stop valve should be classified as Quality Group B, but may be classified as non-seismic Category I if it has been dynamically analyzed to demonstrate structural integrity under SSE loading conditions. However, all pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 apply to this portion of the main steamline from the seismic interface restraint to the turbine stop valve. These requirements are needed to ensure that the quality of the piping material is commensurate with its importance to safety during both operational and accident conditions. In accordance with Position C.1.e of RG 1.29, the turbine stop valve shall be designed to withstand the SSE and maintain its integrity.

The seismic interface restraint must provide a structural barrier between the seismic Category I portion of the main steamline in the reactor building and the non-seismic Category I portions of the main steamline in the turbine building. The seismic interface restraint must be located inside the seismic Category I building. The classification of the main steamline in the turbine building as non-seismic Category I is needed for consistency with the classification of the turbine building. On this basis, the quality and safety requirements imposed on the main steamline from the outermost isolation valve up to, but not including, the turbine stop valve are equivalent to the staff guidelines in Appendix A to SRP Section 3.2.2 and RG 1.29.

The staff and EPRI agree that preventing gross structural failure of the piping and hotwell would provide assurance that leakage from the MSIVs following a design-basis accident would not exceed the 10 CFR Part 100 guideline. In addition, this would ensure the integrity of the main steam drain lines and bypass piping from the first valve to the main condenser hotwell. The remaining issue is the classification of the main steam drain lines and bypass piping between the first normally-closed valve and the condenser hotwell as well as the hotwell itself. <u>The staff proposes that neither the main steam drain and bypass lines from the first valve up to the condenser inlet, nor the piping between the turbine stop valve and the turbine inlet should be classified as safety-related or as seismic Category I. Rather, these lines should be analyzed using a dynamic seismic analysis to demonstrate structural integrity under SSE loading conditions. The turbine stop, control, and bypass valves and the main steam lines from the turbine control valves to the turbine shall meet all of the quality group and quality assurance guidelines specified in SRP Section 3.2.2, Appendix A.</u>

<u>Further, the staff proposes that seismic analyses be performed to ensure that the condenser anchorages and the piping inlet nozzle to the condenser are capable of maintaining their structural integrity during and after the SSE.</u> The dose analysis considers that the condenser is open to the atmosphere. Thus, it is only necessary to ensure that gross structural failure of the condenser will not occur. Similarly, it is only necessary to ensure that failure of non-safety-related SSC resulting from a seismic event will not cause failure of the main steam piping, main steam drain and bypass lines, or condenser.

In its letter of May 13, 1992, the ACRS stated that it agrees with the staff's recommendation for the main steamline classification for both evolutionary and passive BWRs. In a letter dated May 5, 1992, EPRI stated that it agrees with the staff's recommendations and proposed several clarifications, which the staff has incorporated into the above discussion.

The staff concludes that the above-described approach to resolve the BWR main steamline issue for both evolutionary and passive ALWRs provides reasonable assurance that the main steam piping from the outermost isolation valve up to the turbine stop valve, the main steam drain and bypass lines up to the condenser, and the main condenser will retain their pressure and structural integrity during and following a SSE. The staff recommends that the Commission approve the above-described approach to resolve the main steamline classification for both evolutionary and passive ALWRs.

F. Tornado Design Basis

The current NRC regulatory position with regard to design-basis tornados is contained in two documents written in 1974: WASH-1300, "Technical Basis for Interim Regional Tornado Criteria," and RG 1.76 "Design Basis Tornado for Nuclear Power Plants." According to WASH-1300, the probability of occurrence of a tornado that exceeds the design basis tornado (DBT) should be on the order of 10^{-7} per year for each nuclear power plant. The regulatory guide delineates maximum wind speeds of 386 to 597 kilometers per hour (km/hr) (240 to 360 miles per hour (mph)) depending on the regions.

As a result of EPRI's earlier efforts on this EPRI-proposed plant optimization subject, the NRC reevaluated the regulatory positions in RG 1.76 using the considerable quantity of tornado data which has become available since the regulatory guide was developed. The reevaluation is discussed in NUREG/CR-4661 (PNL-9697), "Tornado Climatology of the Contiguous United States," dated May 1986. At the heart of this study is the tornado data tape prepared by the National Severe Storm Forecast Center (NSSFC), which contains data for the 30 year period from 1954 through 1983. This tape contains data for the approximately 30,000 tornados that occurred during the period.

The staff determined that the tornado strike probabilities range from near 10^{-7} per year for much of the western United States to about 10^{-3} per year in the central United States. Based on discussions between the contractor and the staff, wind speed values associated with a tornado having a mean recurrence interval of 10^{-7} per year were estimated to be about 322 km/hr (200 mph) for states west of the Rocky Mountains, and 482 km/hr (300 mph) for states east of the Rocky Mountains.

In its letter of December 6, 1991, EPRI proposed that the design-basis tornado to be used in the design of evolutionary ALWRs be based on a maximum tornado wind speed of 482 km/hr (300 mph) and a tornado strike probability derived from a recurrence interval of 10^{-7} per year. During a meeting with the staff on January 30, 1992, EPRI indicated that it would delete the reference to the tornado recurrence interval from the requirements document. The evolutionary and passive ALWR designers have indicated that their applications will be consistent with the EPRI requirements document.

The tornado design-basis requirements have been used to establish structural requirements (such as minimum concrete wall thicknesses) to protect nuclear plant safety-related SSC against effects not explicitly addressed in review guidance (such as RG or the SRP). Specifically, the staff has routinely reviewed and evaluated aviation crashes (involving general aviation light aircraft), nearby explosions, and explosion debris or missiles, taking into account the tornado protection requirements. The staff's acceptance of EPRI's proposal will necessitate a concurrent review and evaluation of their effect on the protection criteria for some external impact hazards, such as general aviation or nearby explosions. Therefore, external impact hazards will be reviewed on a site-specific basis.

Based on the updated tornado data and the analysis provided in NUREG/CR-4661, the staff concludes that it is acceptable to reduce the tornado design-basis wind speeds to 322 km/hr (200 mph) for states west of the Rocky Mountains, and 482 km/hr (300 mph) for states east of the Rocky Mountains. <u>Therefore, the staff recommends that the Commission approve the position that a maximum tornado wind speed of 482 km/hr (300 mph) be used in the design-basis tornado employed in the design of evolutionary and passive ALWRs</u>. The COL applicant will have to demonstrate that a design capable of withstanding a 482 km/hr (300 mph) tornado will also be sufficient to withstand other site-specific hazards.

In its letter of May 13, 1992, ACRS agreed with the staff that the best available data should be used to establish the tornado design basis. However, ACRS noted the need to account for other potential loads that may previously have been subsumed within the tornado design-basis. The staff's position in the final policy paper has not changed, and the COL applicant will be required to demonstrate that the tornado design envelopes site-specific external impact hazards.

In its letter of May 5, 1992, EPRI agreed with the staff's position to use a 482 km/hr (300 mph) maximum tornado wind speed and to consider other site-specific hazards in the COL or early site permit.

The staff expects that the use of these criteria will not preclude siting the ALWR plant designs on most sites in the United States. However, should an actual site hazard exceed the design envelope in a certain area, the COL applicant would have the option of performing a site specific analysis to verify that the design is still acceptable for that site.

G. Containment Bypass

The phenomenon of containment bypass is associated with either the failure of the containment system to channel fission product releases through the suppression pool, or the failure of passive containment cooling system heat exchanger tubes in the large pools of water outside the containment. The fundamental characteristic of a BWR pressure-suppression containment is that steam released from the reactor coolant system will be condensed (thereby limiting the pressure increase in the containment) and scrubbed of radionuclides in a pool of water (the suppression pool). This is accomplished by directing the steam from the reactor coolant system to the suppression pool through a vent system. However, leakage paths could exist (between the drywell and the wetwell airspace) that could allow steam to bypass the suppression pool and pressurize (or over-pressurize) the containment.

Potential sources of steam bypass include vacuum relief valve leakage, cracks in the drywell structure, and penetrations through the drywell structure. Therefore, the staff concludes that vendors should make reasonable efforts to minimize the possibility of bypass leakage and should account, in their containment designs, for a certain amount of bypass leakage.

In addition, for a containment design that uses an external heat exchanger, the potential exists for containment bypass from a leak in the heat exchanger. High temperatures associated with severe accidents or core debris carried from the reactor vessel could threaten the integrity of the heat exchanger tubes and, therefore, provide a pathway for the release of fission products. Containment sprays in the drywell or wetwell would reduce the effect of suppression pool bypass leakage on containment performance. These systems spray water into the containment and lower its temperature and pressure. They also scrub the containment atmosphere of fission products and mitigate the effects of bypass on fission product distribution. In view of the contribution they can make to accident management, the staff is evaluating the need for containment spray systems for all ALWR designs. The GE ABWR and the ABB-CE System 80+ evolutionary designs have containment spray systems and, therefore, this issue is resolved for evolutionary designs.

In its letter of May 5, 1992, EPRI stated that they believed three separate topics were included in this issue. These topics were (1) interfacing system LOCA; (2) bypass of the suppression pool (BWR); and (3) failure of heat exchanger tubes in the passive containment cooling system (BWR). EPRI also indicated that their requirements documents adequately address these issues.

The staff will address the issue of containment bypass and whether passive designs should contain a containment spray system in a separate Commission paper which will discuss resolution of issues related to source term.

H. Containment Leak Rate Testing

EPRI proposed that the maximum interval between Type C leakage rate tests should be 30 months, rather that the 24-month interval currently required by Appendix J to 10 CFR Part 50. This new maximum interval would apply to both evolutionary and passive plant designs. EPRI proposed this modification to allow some margin between the nominal 24-month refueling interval and the Type C test interval to ensure that plant shutdowns will not be required solely to perform Type C tests. Other issues (such as air lock testing and Type C leak testing methods) have also been raised, but have not been forwarded to the Commission as policy questions.

In parallel with the staff's review of this ALWR issue, the staff has developed proposed changes to Appendix J to 10 CFR Part 50 for all reactors. These changes were transmitted to the Commission in SECY-91-348, "Issuance of Final Revision to Appendix J to 10 CFR 50, and Related Final RG 1.XXX (MS 021-5)." This document proposes modification of the regulation that would allow the increased interval and addresses other issues raised by EPRI. SECY-91-348 also presents the staff's justification for these modifications.

This proposed rulemaking was withdrawn and is currently being reexamined under the Commission-approved program "Elimination of Requirements Marginal to Safety." The staff expects to complete this re-examination and advise the Commission on its final decision about the rulemaking course of action in December 1993.

In addition, the staff has extended (by as much as 1 year) the time interval for performing Type C leakage rate testing on currently operating plants on a case-by-case basis.

In its letter of May 13, 1992, ACRS identified no significant safety penalty caused by this change to the maximum interval between Type C leakage rate tests and agreed with the proposed staff position. In its letter of May 5, 1992, EPRI recommended that the Commission approve the staff's position.

Therefore, the staff recommends that the Commission approve the position that, until the rule change proceedings for Appendix J of 10 CFR Part 50 are completed, the maximum interval between Type C leakage rate tests for both evolutionary and passive plant designs should be 30 months, rather than the 24-month maximum interval currently required in Appendix J to 10 CFR Part 50.

I. Post-Accident Sampling System

10 CFR 50.34(f)(2)(viii) requires the designer to provide the

> capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID-14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole body or 75 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (such as noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations.

RG 1.97 and NUREG-0737, "Clarification of TMI Action Plan Requirements," provide guidance regarding the design of the post-accident sampling system (PASS) used to implement 10 CFR 50.34(f)(2)(viii).

EPRI has proposed deviation from several aspects of the PASS design requirements, as discussed below. EPRI has identified this issue as a plant optimization subject.

Elimination of the Hydrogen Analysis of Containment Atmosphere Samples

EPRI has stated that the hydrogen analysis of the containment atmosphere can be accomplished by the safety-grade containment hydrogen monitor required by 10 CFR 50.34(f)(2)(xvii) and Item II.F.1 of NUREG-0737 "Clarification of TMI Action Plan Requirements." The staff concludes that the safety-grade instrumentation provides adequate capability for monitoring post-accident hydrogen, and this is acceptable justification for an ALWR vendor to use in requesting this deviation. Because this exemption has previously been granted on currently operating plants, the staff does not consider this request to be a policy matter.

In its letter dated May 13, 1992, ACRS agreed that elimination of the hydrogen analysis of containment atmosphere samples is appropriate, given that the safety grade hydrogen monitoring instrumentation will be installed.

Elimination of Dissolved Gas and Chloride Analyses of Reactor Coolant Samples

EPRI considers the analyses of the reactor coolant for dissolved gas and chloride to be unnecessary because venting will remove gases accumulated in the reactor vessel (mainly hydrogen), and because prompt depressurization and cooling will minimize corrosion resulting from the presence of chloride and oxygen. Additionally, the amount of dissolved hydrogen in the reactor coolant can be determined based upon the hydrogen concentration measured in the containment atmosphere.

10 CFR 50.34(f)(2)(viii) and Item II.B.3 of NUREG-0737 specify that the PASS should have the capability to analyze dissolved gases and chloride. This requirement was formulated before reactor vessels were required to have vents. With vented reactor vessels, the information on dissolved gas concentration became less important.

The staff concludes, however, that in PWRs even with vented reactor vessel some postulated accident sequences can occur in which the reactor coolant system is intact at reduced pressure, and heat is removed by subcooled decay heat removal (as in the TMI-2 accident). For these cases, it will not be possible to evaluate concentrations of the dissolved gases in reactor coolant by measuring their concentrations in the containment. For PWRs exposed to these conditions, information on the amounts of dissolved hydrogen in the reactor coolant is an important factor in evaluating post-accident conditions existing in the reactor vessel. The presence of hydrogen can affect flow of coolant in the core. Therefore, the staff concludes that for PWRs the requirement for PASS sampling 24 hours after the accident would be adequate to help ensure long-term decay heat removal.

In BWRs whenever core uncovering is suspected, the reactor vessel is depressurized to within approximately the pressure within the wetwell and the drywell. As a result of this decrease in pressure, dissolved gases will partially pass out of the water phase into the gas phase and under these conditions the concentration of dissolved gases in the reactor coolant would have little meaning. During accidents in which only a small amount of cladding damage has occurred and the reactor vessel has not been depressurized, pressurized reactor water samples may be obtained from the process sampling system. Information on chloride and oxygen concentrations in the reactor coolant, although helpful in ensuring that proper steps are taken to minimize corrosion of reactor components, constitutes a secondary consideration since these samples could be taken at a low pressure. Therefore, for BWRs, there will be no need for taking pressurized samples.

In its letter of May 5, 1992, EPRI stated that sampling of the reactor coolant for dissolved gases and chlorides is not needed because ALWR coolant depressurization systems are highly reliable. Further, EPRI indicated that significant requirements associated with prevention of severe accidents make the need for PASS exceedingly small, especially in view of the limited usefulness to the operators of the information obtained from these measurements. In addition, even if a need for this information develops, EPRI maintained that low levels of fuel failures predicted in ALWRs would permit operators to use routine sampling equipment.

In its letter to ACRS dated June 12, 1992, the staff provided the justification for retaining this sampling equipment for PWRs. The need for the dissolved gas sampling stems from the possibility of partially mitigated severe accidents which do not involve early reactor depressurization, as was demonstrated in the TMI-2 accident. In addition, there is a concern that these reactors may have a problem in maintaining reliable natural circulation and decay heat removal in the presence of non-condensible gases that would evolve during depressurization. This concern is especially important in passive PWRs where decay heat removal systems are highly dependent on natural circulation. Also, in these reactors, non-safety systems would need to be used to perform the final cooldown depressurization. These actions would require that the operator fully appreciate the consequences of depressurizing the plant and possibly introducing non-condensible gases that may interfere with the successful termination of the event.

Determination of chloride concentrations, although helpful in ensuring that plant personnel take appropriate actions to minimize the likelihood of accelerated primary system corrosion following the accident, is a secondary consideration since long term samples could likely be taken at a low pressure. It does not constitute, therefore, mandatory requirement of the PASS.

Therefore, the staff recommends that the Commission approve the position that post-accident sampling systems for evolutionary and passive ALWRs of the pressurized water reactor type be required to have the capability to analyze dissolved gases and chloride, in accordance with the requirements of 10 CFR 50.34(f)(2)(viii) and Item III.B.3 of NUREG-0737. The time for taking these samples can be extended to 24 hours following the accident. For evolutionary and passive ALWRs of the boiling water reactor type, there would be no need for the post-accident sampling system to analyze dissolved gases.

Relaxation in the Time Requirement for Sampling Activity Measurements

EPRI states in the Passive Requirements Document that if boron solution has been added to permit plant shut down, reactor water samples can be taken for boron analyses starting eight hours after the end of power operation. EPRI also states that the samples for activity measurements will not be required until the accident recovery phase.

Item II.B.3 of NUREG-0737 specifies that the PASS should have the capability to obtain coolant and containment atmosphere sampling results within three hours following the accident. The purpose of this requirement is to ensure the capability to draw samples while the accident is still in progress, because analyses of the samples can provide insights for accident mitigation measures. EPRI has committed that the neutron flux monitoring instrumentation will comply with the Category I criteria of RG 1.97. Therefore, this instrumentation will have fully qualified, redundant channels that monitor over the power range of 10^{-6} percent to full power. Based on this commitment, the staff concurs with EPRI's assertion that sampling for boron concentration measurements will not be required for the first 8 hours after an accident.

By contrast, samples for activity measurements are required to evaluate the condition of the core. During the accident management phase, this information will be provided by the containment high-range area radiation monitor, the containment hydrogen monitor, the reactor vessel water level indicator (for BWRs), and the core exit thermocouples (for PWRs). These data will be sufficient to meet the needs of the plant operators for the first 24 hours after an accident. The need for PASS activity measurements will arise during the accident recovery phase when the degree of core damage and general plant contamination will have to be evaluated. Based on this justification, the staff concludes that the requested time extension for sampling activity measurements to 24 hours after an accident is acceptable.

<u>Therefore, the staff recommends that the Commission approve the deviation from the requirements of Item II.B.3 of NUREG-0737 with regard to requirements for sampling reactor coolant for boron concentration and activity measurements using the post-accident sampling system in evolutionary and passive ALWRs.</u>

<u>The modified requirement would require the capability to take boron concentration samples and activity measurements 8 hours and 24 hours, respectively, following the accident.</u>

J. Level of Detail

In its SRM of February 15, 1991, concerning SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," the Commission provided guidance regarding the level of detail of information required to determine the adequacy of design certification applications under 10 CFR Part 52. Although this issue is applicable to all design certification applications, the staff has been reviewing the ABWR as the lead plant in resolving this issue.

The staff identified several areas in the ABWR application where additional information is needed in order to resolve safety concerns. The design detail resulting from the resolution of all of the staff's safety concerns will constitute the level of detail needed to support design certification in accordance with the SRM on SECY-90-377.

The staff has informed the Commission of the progress of its efforts to resolve the level of detail issue and has requested Commission guidance when appropriate. The following Commission papers have addressed issues related to level of detail:

- SECY-91-178, "Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for Design Certification and Combined Licenses;"
- SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews;"
- SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR);"
- SECY-92-214, "Development of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for Design Certification;" and,
- SECY-92-287, "Form and Content for a Design Certification Rule."

The staff concludes that the level of detail issue is applicable to all design certification applications, but expects to resolve this issue in the context of the ABWR review. The discussion provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. The staff is seeking no further guidance from the Commission on this issue.

K. Prototyping

In SECY-91-074, "Prototype Decisions for Advanced Reactor Designs," the staff discussed the process it will use to assess the need for a prototype or other demonstration facility for the advanced reactor designs. The staff stated that it will follow the procedure outlined in SECY-91-074 to identify the various types of testing, up to and including a prototype facility, that may be needed to demonstrate that the advanced reactor designs are sufficiently mature to be certified.

Because the need for prototype testing is a design-specific issue, it cannot be resolved during the EPRI review. The staff has evaluated the submittals of evolutionary ALWRs to assess the need for prototype testing and has not identified any areas that may require such testing.

As discussed in SECY-91-273, "Review of the Vendor's Test Programs to Support the Design Certification of Passive Light-Water Reactors," the need for separate effects and scaled integral testing for passive designs is under consideration. The staff is currently reviewing vendor test programs for the Westinghouse AP600 and the GE SBWR and working closely with the vendors to resolve concerns. Since the issuance of SECY-91-273, the staff has forwarded to the Commission several Commission papers discussing integral system testing requirements and proposed NRC-sponsored confirmatory testing for the AP600 and SBWR designs.

The prototyping discussion provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. If a policy question is identified as a result of its review, the staff will inform the Commission of the issue at the earliest opportunity.

L. ITAAC

ITAAC are required for certified designs in accordance with Subpart B of 10 CFR Part 52. Licensees that reference a certified design will implement the related ITAAC. The Nuclear Management and Resources Council (NUMARC) has designated GE as the industry lead for developing ITAAC on their ABWR application. The staff is working with GE and NUMARC to develop ITAAC for design certification and has kept the Commission apprised of the status of ITAAC. The following ITAAC-related papers have been sent to the Commission:

- SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses," June 12, 1991;
- SECY-91-210, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Requirements for Design Review and Issuance of a Final Design Approval," July 16, 1991;
- SECY-92-053, "Use of Design Acceptance Criteria (DAC) During 10 CFR Part 52 Design Certification Reviews," February 19, 1992;
- SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR)," May 28, 1992;
- SECY-92-214, "Development of Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications," June 11, 1992;
- SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," August 27, 1992; and,
- SECY-92-327, "Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for the GE Advanced Boiling Water Reactor (ABWR)," September 22, 1992.

These papers discuss a wide range of issues related to ITAAC including how non-traditional issues discussed in this paper such as severe accidents and insights from the PRA are included in ITAAC. The discussion provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. The staff will continue to interact with the industry on this matter. Should additional policy questions be raised as a result of the staff's review, the staff will inform the Commission at the earliest opportunity.

M. Reliability Assurance Program

In SECY-89-013, the staff stated that a program called the reliability assurance program (RAP) would be required for design certification to ensure that the design reliability of safety significant SSC is maintained over the life of a plant. In November 1988, the ALWR vendors and EPRI were informed that the staff was considering this matter.

The ALWR RAP would apply to those plant SSCs that are risk-significant (or significant contributors to plant safety) as quantified by the design certification PRA. The purposes of the RAP are (1) to ensure that an ALWR is designed, constructed, and operated in a manner that is consistent with the design assumptions for these risk-significant SSCs, (2) to prevent the reliability of these risk-significant SSCs from degrading during plant operations, (3) to minimize the frequency of transients that challenge ALWR SSCs, and (4) to help ensure that these SSCs function reliably when challenged.

The staff views the RAP for ALWRs as a two-stage program. The first stage applies to the design phase of the plant life cycle, and is referred to as the design reliability assurance program (D-RAP). The second stage applies to the construction and operations phases of the plant life cycle and is referred to as the operational reliability assurance program (O-RAP). An applicant for design certification shall be required to establish the framework for the RAP (e.g., scope, purpose, objective, and essential elements of an effective RAP) and shall implement those portions of the D-RAP that apply to design certification. A COL applicant would augment the design certification D-RAP with site-specific design information and would implement the balance of the D-RAP, including input to the procurement process. The COL applicant would also establish and implement the O-RAP.

The O-RAP can be thought of as an umbrella program that would integrate aspects of existing programs (e.g., maintenance, surveillance testing, inservice inspection, inservice testing, and QA) to achieve its objective. The O-RAP would apply to the construction and operation phases of plant life. To ensure regulatory coherence between RAP and the maintenance rule (10 CFR 50.65), performance goals for risk-significant SSCs would be established under the O-RAP based on input from the D-RAP. As such, performance and condition monitoring requirements for maintaining the reliability of risk-significant SSCs would be established. In addition, O-RAP would provide a feedback mechanism for periodically re-evaluating risk significance based on actual equipment, train, or system performance. The majority of the O-RAP would be based on the requirements of 10 CFR 50.65.

The staff has completed its review of the RAP in the EPRI URD for the evolutionary ALWR, as documented in the final safety evaluation report (FSER) for Chapter 1, Section 6. The staff has also documented the results of its review of the GE ABWR D-RAP in the draft final safety evaluation report (DFSER) for Chapter 17.3. The staff has had the benefit of discussions with the ACRS regarding the form and content of the ALWR RAP. In its letter dated

October 15, 1992, concerning proposed guidance for implementation of the maintenance rule (10 CFR 50.65), the ACRS noted the similarity between the maintenance rule, the license renewal rule, and the RAP. The ACRS stated that consistent staff guidance is needed on the elements of an acceptable program that will satisfy these three sets of requirements. The staff has incorporated the ACRS comments in developing its position on the RAP.

The following section summarizes the position the staff has taken in performing its review of D-RAP submittals contained in the design certification applications and the EPRI URD for the evolutionary and passive ALWRs.

Interim Staff Position on RAP

For design certification, the staff position is that a high level commitment to a RAP applicable to design certification (D-RAP) should be required as a non-system generic Tier 1 requirement with no associated ITAAC. The details of the D-RAP, including the conceptual framework, program structure, and essential elements, should be provided in the SSAR. The SSAR for the D-RAP should also (1) identify and prioritize a list of risk-significant SSCs based on the design certification PRA and other sources, (2) ensure that the vendor's design organization determines that significant design assumptions, such as equipment reliability and unavailability, are realistic and achievable, (3) provide input to the procurement process for obtaining equipment that satisfies the design reliability assumptions, and (4) provide these design assumptions as input to the COL for consideration in the operational reliability assurance program (O-RAP). A COL applicant would augment the design certification D-RAP with site-specific design information and would implement the balance of the D-RAP, including input to the procurement process. The COL applicant would also establish and implement the O-RAP. The staff will review the COL applicant's D-RAP and O-RAP as part of the COL application.

The staff position on O-RAP will be provided in a future Commission paper. In developing its position on O-RAP, the staff will ensure that regulatory coherence exists between O-RAP and the requirements of the maintenance rule and the license renewal rule.

The staff's proposed resolution of the RAP for design certification will be provided to the Commission in a separate Commission paper which will discuss the regulatory treatment of active non-safety systems in passive plants. The discussion provided in this section is for information only and is provided to identify a complete list of issues applicable to evolutionary and passive designs. The staff will continue to interact with the industry on this matter. Should additional policy questions be raised as a result of the staff's review, the staff will inform the Commission at the earliest opportunity.

N. Site-Specific Probabilistic Risk Assessments and Analysis of External Events

In its "Policy Statement on Severe Accidents Regarding Future Designs and Existing Plants," issued on August 8, 1985 (50 FR 32138), the Commission stated that applicants for future evolutionary reactor plant design reviews should complete a PRA. The Commission also stated that applicants should consider improving the means to avoid or mitigate severe accident vulnerabilities exposed by the PRA in order to help ensure the public health and safety. Further, the Commission stated that evolutionary plant vendors should use the PRA in considering a range and combination of alternatives that address unresolved and generic issues and to search for cost-effective means to reduce the risk associated with severe accidents. In the policy statement, the Commission directed the staff to review evolutionary ALWR designs to determine the safety acceptability of the design, stressing deterministic engineering analysis and judgment, complimented by PRA. After issuing this policy statement, the Commission promulgated 10 CFR Part 52. Section 52.47 requires that an application for design certification contain a design-specific PRA.

In GL 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," and its supplements, the staff stated that construction permit holders and power reactor licensees should consider the safety implications of both internal and external events. Such consideration should involve performing separate individual plant examinations (IPEs) and individual plant examinations for external events (IPEEE). PRAs and IPEs that have evaluated both internal and external events generally estimate the risks from external events to be the same order of magnitude as internal events. Therefore, the staff concludes that the design-specific PRAs required in 10 CFR 52.47 should include an assessment of both internal and external events.

Lessons from past risk-based studies indicate that fire, internal floods, and seismic events can be important potential contributors to core damage. However, the estimates of core damage frequencies for fire and seismic events continue to include considerable uncertainty. Consequently, the staff concludes that fire and seismic events can best be evaluated using simplified probabilistic methods and margins methods similar to those developed for existing plants, supported by insights from internal event PRAs (including ALWR design-specific PRAs). The designer should use traditional probabilistic techniques to study internal floods.

Fire events can be evaluated using simplified methods such as EPRI's Fire Induced Vulnerability Evaluation (FIVE) methodology rather than full scale PRAs. Ascribing to these methods, the designer would focus on the capacity of the design to withstand the effect of fire, using qualitative and quantitative methods rather than a strictly quantitative PRA fire analysis.

The staff concludes that the plant designer can best determine the seismic capability of the plant through a combined approach that takes advantage of the strengths of both PRA and margins methods. This approach (based on an internal events PRA, its existing event and fault trees, and its random failures and human errors) allows for a comprehensive and integrated treatment of the plant's response to an earthquake. This approach should yield meaningful measures of a proposed design's seismic capability.

The major difference between a seismic PRA and the proposed PRA-based margins approach is that the latter does not convolute fragility curves with hazard curves. Rather, the PRA-based margins approach measures the robustness of the plant to withstand earthquakes of a given g-level. This method eliminates the need to deal with uncertainty in the seismic hazard curve for the site and identifies potential design-specific seismic vulnerabilities. Understanding these vulnerabilities may be useful in developing the reliability assurance programs, identifying operator training requirements, and focusing on accident management capabilities.

The staff will require each plant designer to perform a PRA-based margins analysis to identify the vulnerabilities of their design to seismic events. The plant designer should construct plant logic models covering the various systems that could be used to prevent core damage. Typically, this would be accomplished by modifying design-specific PRA models for internal events to include logic important in considering seismic failures. The models would not include data from site-specific or generic seismic hazard curves. The designer would then determine all important accident sequences using the event trees and fault trees (based on fragility data for each component for each sequence). In addition, the designer would determine the value of the minimum high confidence, low probability of failure (HCLPF) for the plant by determining the HCLPF values for the important SSCs for each accident sequences. The HCLPF values calculated in this manner can be used to measure the plant's robustness and to provide an acceptable estimate of the earthquake ground motion which the plant is expected to be able to survive without core damage. In general, the value of the plant HCLPF should be at least twice the design ground motion zero period acceleration. If not, the designer should perform a more detailed evaluation to identify any vulnerability against which the plant requires strengthened protection. HCLPF calculations also indicate which components and systems limit the seismic capability of the plant.

In its letter of May 8, 1992, EPRI stated that they were unaware of suitable margins approaches to evaluate external events other than a seismic margins assessment. In its letter of August 21, 1992, EPRI provided additional comments regarding the staff's proposed requirements for analysis of certain internal and external events. EPRI concluded that their requirements for separation of redundant SSE by physical barriers designed to withstand full-height compartment flooding should make core damage events resulting from internal flooding very unlikely. In addition, EPRI concluded that the proposed requirements for analyses of internal fire and seismic events would be addressed satisfactorily by their requirements documents, except for the provision of a seismic margin value twice the magnitude of the SSE. EPRI recommended that the seismic margin value be changed to 1.5 times the magnitude of the SSE to be consistent with operating plants. In addition, EPRI noted that the as-built verification phase of the seismic margins evaluations can be based upon applying the appropriate margin to the site-specific SSE.

The staff concludes that separated safety divisions do not necessarily preclude internal floods from being significant contributors to risk. The staff notes that evolutionary ALWR designs that include physical separation between safety divisions appear to respond better to internal fires and floods than do traditional designs. However, although this physical separation should reduce the expected frequency and severity of these events, it does not preclude them from being important contributors to risk. For example, an internal flooding PRA study for an evolutionary design has already identified several design improvements that are needed to achieve the desired level of internal flood protection. The staff believes that the systematic evaluation required during the development of an internal flooding PRA is necessary to ensure that potential flood vulnerabilities are identified.
The EPRI seismic margins methodology limits the search for success paths to two, and will not provide some important insights concerning human error and random failure. Therefore, this methodology does not appear adequate for evaluating evolutionary and passive ALWR designs. In addition, the EPRI margins method is limited in its useful application for passive plant margins analyses as is the NRC margins method used in the IPEEE, since these margins approaches are based on insights from PRAs of operating plants (PWRs and BWRs). The staff concludes that it would be inappropriate to rely exclusively on either of these methods to perform seismic margins analyses for passive designs.

In addition, the staff concludes that a well-designed plant should have a plant HCLPF at least twice the magnitude of its SSE. Therefore, if the PRA-based margins analysis only looks at accelerations up to 1.5 times the magnitude of the SSE, the analysis could effectively screen out potential design-specific seismic vulnerabilities. The staff would anticipate that analysis of a fully developed seismic PRA will identify seismic events as significant contributors to risk (perhaps the largest, given the significant estimated reduction in internal event core-damage frequency in evolutionary ALWRs). Therefore, it is important to fully understand potentially significant seismic vulnerabilities and other seismic insights. This information would be captured by a PRA-based seismic margins analysis that considers sequence-level HCLPFs and fragilities for all sequences leading to core damage or containment failures up to approximately twice the magnitude of the SSE.

Details of the specific site characteristics will likely not be available until the COL review stage. The staff intends to require a COL applicant to perform a site-specific PRA that addresses all applicable site-specific hazards (such as river flooding, storm surge, tsunami, volcanism, or hurricanes). The staff will review the site-specific PRA to ensure that no vulnerabilities are introduced by siting the standardized plant at a location where external hazards could pose an unacceptable or unanticipated risk.

In its letter of May 13, 1992, and in discussions with the staff, ACRS noted that the staff, an ALWR vendor, and a COL applicant may experience significant obstacles if design vulnerabilities from site-specific external events are discovered at the COL review stage. ACRS requested more information on how the staff proposes to deal with unacceptable findings, resulting from the site-specific PRA, identified during the COL application review process.

In SECY-92-287, "Form and Content for a Design Certification Rule," the staff proposed the appropriate processes to modify Tier 1 and Tier 2 design certification information. For example, if a site-specific PRA identifies a serious generic design flaw that meets the "adequate protection" threshold, the NRC can initiate rulemaking to amend the design certification rule. If the site-specific PRA identifies a site-specific design weakness, the COL applicant will have the option to request an exemption to the design certification rule to correct the deficiency.

However, the staff will require that ALWR vendors perform bounding analyses of site-specific external events likely to be a challenge to a plant. When a site is chosen, its particular siting characteristics can then be compared to those of the bounding analyses in order to minimize the potential for the site-specific PRA to identify significant site-specific weaknesses for the standard design. Before certifying the design, the staff will evaluate fires, internal floods, and other external events that are not site dependent and will evaluate submitted bounding analyses for site-specific external events.

<u>Therefore, the staff recommends that the Commission approve the position that the analyses submitted in accordance with 10 CFR 52.47 should include an assessment of internal and external events. PRA insights will be used to support a margins-type assessment of seismic events. A PRA-based seismic margins analysis will consider sequence-level HCLPFs and fragilities for all sequences leading to core damage or containment failures up to approximately twice the magnitude of the SSE. Simplified probabilistic methods, such as EPRI's FIVE methodology, will be used to evaluate fires. Traditional probabilistic techniques should be used to evaluate internal floods.</u>

<u>Secondly, the staff recommends that the Commission approve the position that ALWR vendors should perform bounding analyses of site-specific external events likely to be a challenge to the plant (such as river flooding, storm surge, tsunami, volcanism, high winds, and hurricanes). When a site is chosen, its characteristics should be compared to those assumed in the bounding analyses to ensure that the site is enveloped. If the site is enveloped, the COL applicant need not perform further PRA evaluations for these external events. The COL applicant should perform site-specific PRA evaluations to address any site-specific hazards for which a bounding analysis was not performed or which are not enveloped by the bounding analyses to ensure that no vulnerabilities due to siting exist.</u>

The COL applicant should submit the results of the comparison of the bounding analyses to the site characteristics and any site-specific PRA information to the NRC at the COL review stage. For the GE ABWR and the ABB-CE System 80+

designs, the staff will work with the vendors to ensure a fundamental understanding of the potential vulnerabilities of their designs to external events. In order to maintain the design certification review progress, the staff will encourage, but not require, the vendors to complete these bounding analyses before issuance of a design certification rule. If the vendors choose not to perform the bounding analyses, they run a risk that site-specific hazards may limit acceptable sites for the design. For passive ALWR vendors who have submitted an application for design approval to the NRC, the staff will require that these bounding analyses be submitted and reviewed by the staff before issuance of final design approval. For subsequent design approval of evolutionary and passive ALWRs, the staff will require that the applicant submit bounding analyses for external events with the application.

At the COL stage, the staff will review the site-specific characteristics to ensure that events enveloped by bounding analyses at the design stage or evaluated by a site-specific PRA have been properly addressed. The staff plans to conduct walk-down inspections to confirm that design commitments have been met.

O. Severe Accident Mitigation Design Alternatives

As discussed in SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," dated July 31, 1991, the National Environmental Policy Act (NEPA), Section 102(C)(iii), requires, in part, that

> ...all agencies of the Federal Government shall...(C) include in every recommendation or report on proposals for legislation and other major Federal actions significantly affecting the quality of the human environment, a detailed statement by the responsible official on...(iii) alternatives to the proposed action.

In Limerick Ecology Action v. NRC, 869 F.2d 719 (3rd Cir. 1989), the U.S. Court of Appeals effectively required the NRC to include consideration of severe accident mitigation design alternatives (SAMDAs) in the environmental impact review performed as part of the operating license application for the Limerick Generation Station. A NEPA evaluation, in the form of an environmental impact statement that includes consideration of SAMDAs, is an essential element of an application for a COL under Subpart C of 10 CFR Part 52, for those applications that reference a design certified under Subpart B. In SECY-91-229, the staff presented several options concerning the treatment of SAMDA issues as they related to the certification of standard plant designs. The staff recommended that the Commission approve the following recommendations:

- address SAMDAs for certified designs in a single rulemaking;
- approve the staff's approach for considering the costs and benefits of the review of SAMDAs for standard plant design certification; and,

- approve the staff's proposal to advise applicants for a final design approval and design certification that they must assess SAMDAs and provide the rationale supporting their decisions.

In its SRM of October 25, 1991, the Commission approved the staff's recommendations and requested that they be kept informed on the staff's progress in evaluating the SAMDAs for final design approval and design certification applications.

Consistent with the third recommendation of SECY-91-229, the staff requested that ALWR vendors assess SAMDAs for their designs. This assessment and subsequent staff review is in addition to the safety consideration of severe accident issues discussed in this paper. The staff is currently addressing responses to this request in the context of the ABWR and System 80+ reviews.

The discussion provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. The staff will continue to interact with the industry on this matter. Should additional policy questions be raised as a result of the staff's review, the staff will inform the Commission at the earliest opportunity.

P. Generic Rulemaking Related to Design Certification

In SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light-Water Reactor (LWR) Designs," dated August 16, 1991, the staff provided the Commission with recommendations regarding generic rulemaking related to design certification. The staff recommended that the Commission:

- approve the staff's proposal to proceed with design-specific rulemaking through individual design certifications to resolve selected technical and severe accident issues for the ABWR and System 80+ designs; and,

- note the staff's intent to proceed with generic rulemaking, where appropriate for evolutionary and passive designs, as information becomes available from ongoing efforts on these issues, independent of the design review and certification processes.

The staff has not yet received Commission guidance on SECY-91-262.

As discussed in SECY-91-262, the staff concludes that consideration of generic rulemaking in lieu of design-specific rulemaking is applicable to all final design approval and design certification applications. However, the design of the passive plants is not sufficiently developed at this time for the staff to determine whether generic rulemaking should be initiated for passive plant designs. Certain generic rulemaking activities related to the evaluation of source terms during postulated severe accidents are ongoing, and the results of these rulemakings may be used during design certification of the passive reactor designs. Currently, this work is focused on updating 10 CFR Part 100 to separate siting criteria from reactor design criteria. The staff plans to incorporate the revised source-term criteria in 10 CFR Part 50. In addition, the staff plans to consider the incorporation of generic severe accident criteria. The discussion provided in this section is for information only, and is provided to identify a complete list of issues applicable to the passive designs. Should additional policy questions be raised as a result of the staff's review, the staff will inform the Commission at the earliest opportunity.

Q. Defense Against Common-Mode Failures in Digital Instrumentation and Control Systems

Instrumentation and control (I&C) systems help ensure that the plant operates safely and reliably by monitoring, controlling, and protecting critical plant equipment and processes. The digital I&C systems for ALWRs differ significantly from the analog systems used in operating nuclear power plants. Specifically, digital I&C systems share more data transmission functions and shares more process equipment than their analog counterparts.

Redundant trains of digital I&C systems may share data bases (software) and process equipment (hardware). Therefore, a hardware design error, software design error, or software programming error may result in a common-mode or common-cause failure of redundant equipment. The staff is concerned that the use of digital computer technology in I&C systems could result in safety-significant common-mode failures. The NRC staff developed these concerns in SECY-91-292, "Digital Computer Systems for Advanced Light Water Reactors." Some of the major points in that paper are summarized as follows:

- Common mode failures could defeat the redundancy achieved by the hardware architectural structure, and could result in the loss of more than one echelon of defense-in-depth provided by the monitoring, control, reactor protection, and engineered safety functions performed by the digital I&C systems.

- The two principal factors for defense against common-mode and common-cause failures are quality and diversity. Maintaining high quality will increase the reliability of both individual components and complete systems. Diversity in assigned functions (for both equipment and human activities) equipment, hardware, and software, can reduce the probability that a common-mode failure will propagate.

- The staff intends to require some level of diversity, such as a reliable analog backup.

Current regulations applicable to analog I&C systems also apply to digital I&C systems. In addition, the staff has developed limited guidance for digital I&C systems in RG 1.152, "Criteria for Programmable Digital Computer Software Systems in Safety-Related Systems of Nuclear Power Plants." However, as discussed in SECY-91-292, there are currently no regulatory requirements that adequately address the potential safety concerns associated with digital I&C systems.

Quality and diversity are important defenses against common-mode failures. However, there are no standards for certifying the design of digital I&C systems for application in nuclear power plants. In Enclosure 2 of SECY-91-292, the staff discussed regulatory requirements that it is considering to help ensure defense against common-mode failures, including the following areas:

- assessment of diversity;
- engineering activities;
- design implementation; and,
- safety classification of I&C systems.

The staff has made significant progress in establishing regulatory guidance that could be used to assess diversity. With the support of LLNL the staff has performed a study of the GE ABWR design to assess the adequacy of its defense-in-depth and diversity. This assessment was performed using the method described in NUREG-0493, "A Defense-in-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," for each transient and accident evaluated in Chapter 15 of the ABWR safety analysis report. The staff is using the results of this assessment to help determine the additional diversity necessary to defend against postulated common-mode software and hardware failures in the ABWR.

EPRI discussed requirements for engineering activities and design implementation for digital I&C systems in Chapter 10 of the EPRI ALWR for both evolutionary and passive plants. In SECY-91-292, the staff discussed the role of the EPRI URD as providing a frame of reference for the development of acceptance criteria for the design to adequately satisfy the requirements in the URD. The criteria needed to satisfy the requirements for engineering activities and design implementation would be developed by the staff using applicable national and international standards where available, or expert opinions where adequate standards have not been developed, during the review of specific ALWR designs.

As discussed in SECY-91-292, the staff is continuing to develop safety classification criteria for I&C systems in ALWR designs. The international technical community, through the International Electrotechnical Commission (IEC), recently issued an IEC standard, "The Classification of Instrumentation and Control Systems Important to Safety for Nuclear Power Plants." EPRI proposed certain classification positions in its "ALWR Position Paper for Passive System Classification and Requirements," submitted by a letter dated March 19, 1992. The staff is considering both of these documents before reaching a final position on safety classification criteria for I&C systems in passive ALWR designs. The safety classification of digital I&C systems relates to diversity through the defense-in-depth assessment by crediting systems that have previously been classified as non-safety systems.

Recently, increased attention has been given to detailed assessments of the integrity of software applied to safety-critical functions. These assessments have covered a broad range of applications, including computer-based medical treatment facilities, computer-based fly-by-wire aircraft control systems, and nuclear power plant protection systems. The staff found a consensus among computer science and software engineering experts that such safety-critical applications should be backed-up by some system not based on software. The experts based this opinion on the facts that the quantitative estimate for the reliability of I&C systems based on high-integrity software cannot yet easily be determined. The type of this backup and the functions it should perform depend on the specific equipment and design features of the I&C system.

The EPRI ALWR requirements document places special emphasis on common-mode failures to ensure they are addressed in man-machine interface system (M-MIS) designs. EPRI stated that the ALWR Program has recognized from the onset that there is currently no accepted standard to accurately quantify software reliability. To offset this concern, the ALWR Program has emphasized the need for software quality and for a defense-in-depth approach to ensure the integrity of I&C functions including requirements for a backup hardwired manual actuation capability for system-level actuation of safety functions.

As previously discussed, the staff has established potential regulatory guidance to ensure adequate diversity for digital I&C system applications. As a result of its review, the staff proposed an approach, in the draft Commission paper dated June 25, 1992, for assessing the defenses against common-mode failures in a design. The proposed approach also specified requirements for a backup system which is not based on software and which is used for system-level actuation of critical safety functions and displays of safety parameters.

After carefully reviewing ACRS, industry, and vendor comments, the staff has developed a final position. The staff has concluded that analyses that demonstrate adequate, rather than equivalent, defense against the postulated common-mode failures would be allowed in the diversity assessment required of the applicant. The critical safety functions that require backup manual controls and displays would be specified. The staff would consider allowing more flexibility in implementing the requirements for an independent set of displays and controls. The necessary degree of flexibility depends on the specific equipment and design features of the I&C system and will be evaluated for each design. The intent is to permit the use of diverse digital equipment that is not affected by the identified common-mode failures and to reduce complexity in the design. The staff will not require only analog equipment and will consider allowing simple digital equipment.

As a result of these changes, the staff revised the initial position proposed in the draft Commission paper. The staff recommends that the Commission approve the following revised staff position:

1. <u>The applicant shall assess the defense-in-depth and diversity of the proposed instrumentation and control system to demonstrate that vulnerabilities to common-mode failures have adequately been addressed.</u> The staff considers software design errors to be credible common-mode failures that must specifically be included in the evaluation. An acceptable method of performing analyses is described in NUREG-0493, "A Defense-In-Depth and Diversity Assessment of the RESAR-414 Integrated Protection System," March 1979. Other methods proposed by an applicant will be reviewed individually.
2. <u>In performing the assessment, the vendor or applicant shall analyze each postulated common-mode failure for each event that is evaluated in the accident analysis section of the safety analysis report (SAR). The vendor or applicant shall demonstrate adequate diversity within the design for each of these events.</u> For events postulated in the plant SAR, an acceptable plant response should not result in a non-coolable geometry of the core, violation of the integrity of the primary coolant pressure boundary, or violation of the integrity of the containment.

3. <u>If a postulated common-mode failure could disable a safety function, then a diverse means, with a documented basis that the diverse means is unlikely to be subject to the same common-mode failure, shall be required to perform either the same function or a different function.</u> The diverse or different function may be performed by a non-safety system if the system is of sufficient quality to perform the necessary function under the associated event conditions. Diverse digital or non-digital systems are considered acceptable means. Manual actions from the control room are acceptable if adequate time and information are available to the operators. The amount and types of diversity may vary among designs and will be evaluated individually.

4. <u>A set of safety-grade displays and controls located in the main control room shall be provided for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. The displays and controls shall be independent and diverse from the safety computer system identified in items 1 and 3 above.</u> The specific set of equipment shall be evaluated individually, but shall be sufficient to monitor the plant states and actuate systems required by the control room operators to place the nuclear plant in a hot-shutdown condition. In addition, the specific equipment should be intended to control the following critical safety functions: reactivity control, core heat removal, reactor coolant inventory, containment isolation, and containment integrity.

   The displays and controls shall be hardwired in the safety computer system architecture to the lowest practical level. To achieve system-level actuation at the lowest level in the safety computer system architecture, the controls may be hardwired either to analog components or to simple, dedicated, and diverse software-based digital equipment that performs the system-level actuation logic. The safety parameter displays may include digital components exclusively dedicated to displays. This requirement would provide for an independent and diverse control logic for manual, system-level actuation of the safety function that would be connected downstream of the lowest-level safety software-based component without affecting the hardware (interconnecting cables and interfaces) between the lowest-level electronic cabinets and the plant's electromechanical equipment.

Human engineering principles and criteria shall be applied to the selection and design of the particular displays and controls. The design of the displays and controls shall ensure that the human system interface shall be adequate to support the human performance requirements.

Hardwired, system-level controls and displays provide the plant operators with unambiguous information and control capabilities. These controls and displays are required to be in the main control room to enable the operators to expeditiously mitigate the effects of the postulated common-mode software failure of the digital safety I&C system. The control room would be the center of activities to safely cope with the event, which could also involve the initiation and implementation of the plant emergency plan. The design of the plant should not require operators to leave the control room for such an event. For the longer term recovery operations, credit may be taken for actions from outside the main control room, when the emergency response organization is fully briefed and in place to take such actions.

R. Steam Generator Tube Ruptures

The staff has identified two distinct issues related to steam generator tube ruptures (SGTRs). These issues, involving multiple ruptures specific to passive PWRs and containment bypass potential resulting from SGTRs, are discussed below.

Multiple Steam Generator Tube Ruptures for Passive PWRs

A design-basis accident involving SGTR in the current generation of pressurized water reactors is a rupture of one steam generator (SG) tube, with a rate of discharge of primary coolant through the SG tube break greater than the normal charging capacity of the reactor coolant inventory control system. SRP Section 15.6.3 requires the applicant to conduct an analysis for a single SGTR, but there is currently no requirement to perform an analysis for multiple SGTRs. The staff is considering whether multiple SGTRs should be included in the plant design basis for advanced PWRs.

In NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety

Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," dated September 1988, the staff estimated the probabilities of single and multiple tube ruptures. When the staff prepared these estimates in 1986, four single SGTRs had occurred in PWRs. All occurred in the United States and all the affected plants were Westinghouse plants: Point Beach Unit 1 (February 1975); Surry Unit 2 (September 1976); Prairie Island Unit 1 (October 1979), and R.E. Ginna (January 1982). Since that time, two more single SGTRs have occurred in the U.S., one at North Anna Unit 1 (July 1987) and another at McGuire Unit 1 (March 1989).

In NUREG-0844, the staff estimated the frequency of a single SGTR to be 1.5×10^{-2} per reactor year (RY). The staff based this estimate on the four events that occurred in approximately 300 "mature" reactor-years of Westinghouse plant operation in the U.S. ("mature" RYs are accumulated after the first 2 years of plant operation). ABB-CE and Babcock and Wilcox plants, which at that time had accumulated 77 and 66 mature RYs, respectively, without experiencing any SGTRs, were assumed to have the same probability of SGTRs as Westinghouse plants. In the same report, the staff estimated the probability of a multiple tube rupture event, using binomial statistics, as 1.6×10^{-3} per RY. The staff based this estimate on a 50-percent level of confidence (probability) for an event that had never occurred in the approximately 440 mature RYs accumulated among all U.S. PWRs at that time.

Since the staff issued NUREG-0844, the total number of mature RYs of operation for both Westinghouse PWRs and all U.S. PWRs has approximately doubled. Westinghouse plants have now accumulated approximately 535 mature RYs, and all U.S. PWRs have accumulated about 827 mature RYs. Experience with Westinghouse plants (6 SGTRs in 535 RYs) indicates that the frequency of a single SGTR is approximately 1.1×10^{-2} per RY. ABB-CE and Babcock and Wilcox plants appear to have lower SGTR frequencies. With a failure rate of 1.1×10^{-2} per RY, about 3 SGTRs should have statistically resulted in about 300 RYs of operating experience, and none have occurred. The NRC has not received any report of an multiple SGTR in any U.S. or foreign plant. For consistency with the NUREG-0844 estimate of the probability of multiple SGTR events, the staff has derived a new estimate based on a 50 percent confidence level for an event that has not occurred in approximately 827 RYs of U.S. PWR operation to date. This estimated frequency for an multiple SGTR event is approximately 8.4×10^{-4}/RY.

SGTRs are generally grouped into two categories: those which occur as initiating events, and those which occur as a consequence of other events that increase the stress on the tubes. The probability estimates given above are for SGTRs that occur as initiating events. These events include random SGTRs caused by degradation of the tube over time, and SGTRs caused by or associated with damage from foreign objects that may be present in the steam generator. Of the four SGTRs reported in NUREG-0844, two (Ginna and Prairie Island) are believed to have been caused at least in part by the impact of foreign objects on the steam generator tubes. The SG tubes in other plants have also leaked because of damage from foreign objects, although this leakage did not exceed the makeup capacity of reactor coolant inventory control systems. This issue is of concern in the context of determining the credibility of multiple SGTR events. While it would seem highly improbable that two random SGTR failures would occur simultaneously (as indicated in NUREG-0844), damage or tube failure caused by a foreign object could be a more likely initiator of an multiple SGTR. In the Ginna event, the licensee examined the SG tubes after the event and found that although only 1 SG tube had ruptured, more than 20 had been severely damaged.

The staff is reviewing the issue of whether to consider a single SGTR or an multiple SGTR as the design-basis event for the AP600. The staff is concerned that an AP600 plant could respond in substantially different ways to the two accidents, and that an multiple SGTR event could pose substantial challenges to the plant's passive safety systems.

In dealing with an SGTR in a conventional plant, operators isolate the faulted SG and reduce the primary system pressure to help stop primary-to-secondary leakage. The operators use the safety-related pressure- and inventory-control systems in these plants (pressurizer spray, high-pressure safety injection) to carry out these procedures. While no multiple tube ruptures have occurred, leakage rate would likely increase with the number of tubes ruptured, and the operators would be required to act to mitigate the consequences of the event as quickly as possible. However, the basic procedures to be employed by plant operators in such an event would be similar to those used for single tube ruptures, and the plant conditions would probably be similar during the transient to those in a single SGTR. Analyses and tests of multiple tube rupture at the SEMISCALE facility have confirmed that the basic plant response is also similar to that for a single SGTR event.

The AP600 plant includes no active safety-related inventory- or pressure-control systems. The core makeup tanks (CMTs) add high-pressure inventory by providing a gravity-driven injection of borated water. A natural circulation passive residual heat removal (RHR) system provides safety-related decay heat removal. The AP600 also uses an automatic depressurization system (ADS) to reduce the primary system pressure in the event of a LOCA and this ADS permits injection of a large amount of low-pressure makeup water from the incontainment refueling water storage tank (IRWST). The first stage of the ADS is triggered upon reducing the CMT level to a predetermined point, with subsequent stages actuated as the CMT level decreases.

The pressure of the primary system should be reduced to about that of the secondary system to inhibit primary-to-secondary leakage. However, primary system depressurization below that of the secondary system appears undesirable during an SGTR. Using the ADS will likely further lower the RCS pressure to such an extent that unborated water could flow from the secondary side of the SGs back into the primary system. This could cause reactivity to increase in the core, with possible detrimental results.

Westinghouse representatives assert that the AP600 has been designed with sufficient margin to the ADS initiation setpoint to allow at least 30 minutes of CMT injection after a single SGTR without triggering the ADS. Westinghouse representatives indicate that this should be sufficient time for the operators to employ both safety-related and available non-safety-related systems to reduce RCS pressure, isolate the faulted SG, and terminate the event.

However, if an multiple SGTR occurs, with a substantially greater leakage rate of primary coolant, the AP600 might not be able to accommodate the accident without actuating the ADS. The operators will have substantially less time to bring the event under control before the CMT level is reduced to the ADS setpoint. ADS actuation might then result in secondary-to-primary leakage of unborated water. This water could flash to steam as it enters the RCS if the SG water is above the saturation temperature at the primary system pressure. Since the passive safety systems of the AP600 rely on small differential pressures to circulate and inject emergency core coolant (ECC), introducing a large amount of steam into the RCS from flashing secondary water could disrupt or degrade ECC injection. Therefore, contrary to the response of current plants, the plant may respond to an multiple SGTR event in a manner considerably different than that for a single SGTR. The consequences may also differ significantly.

The designer could provide a number of methods to minimize the consequences of multiple SGTRs, especially to retard or prevent secondary-to-primary leakage or to lessen the amount of reactivity added as a result. These methods include (1) depressurizing the secondary system to maintain the RCS pressure at a value greater than the secondary pressure and prevent back leakage; (2) providing a system to borate the secondary water automatically if it leaks into the RCS from the steam generator; (3) or providing procedures that inhibit ADS actuation if the primary-to-secondary barrier is breached. The staff is not aware that Westinghouse is considering any of these SGTR mitigation methods for the AP600.

Design-basis accidents, such as large- and small-break LOCAs, have estimated frequencies of occurrence on the order of 10^{-3} to 10^{-5} per RY. These events generally provide the most rigorous test of the plant's safety systems. The staff recognizes, however, that the multiple SGTR frequency could be in the range of 10^{-3} to 10^{-4} per RY and that passive plant response for a multiple SGTR could significantly differ from that for a single tube rupture. This recognition has led the staff to conclude that rupture of more than a single tube should be considered within the design basis of the plant.

The staff is continuing to evaluate the appropriate number of ruptured SG tubes that should be included in the design basis for the AP600. As a minimum, the plant designer or applicant should analyze the single and multiple SGTR events to determine, to the extent possible, the quantitative differences in the plant's response to these events. Therefore, as an interim step, the staff intends to require that Westinghouse analyze multiple ruptures of two to five tubes in its AP600 safety analysis. The staff will report to the Commission when it determines the number of tube ruptures to be incorporated into the design basis of the AP600.

EPRI stated that their URD require that future pressurized water reactors have substantially improved capability to handle SGTRs. EPRI also stated that these requirements address material, design, and operation improvements to prevent SGTRs, as well as design features to improve the performance and response of the plant after an SGTR. EPRI has also concluded that passive plants are not unique with regard to multiple SGTRs, and accordingly, multiple SGTRs should not be included in the design basis for passive LWRs.

In its letter of September 16, the ACRS stated that it agreed with the staff's recommendation that Westinghouse should analyze the AP600 response to ruptures of up to five SG tubes and requested that the staff provide a "better technical basis" for estimating the frequency of multiple SGTRs.

In its letter to ACRS of October 22, 1992, the staff noted that the calculation for estimating the frequency of multiple SGTRs was not meant to represent a rigorous statistical analysis of multiple SGTR frequency. Rather, this calculation was intended to show that the approximate frequency of multiple SGTRs of a few tubes is on the same order of magnitude as that of other limiting faults. The frequency calculation, coupled with the unique response of the passive safety systems to accidents involving loss of primary inventory, has led the staff to conclude that evaluation of multiple SGTRs up to about five tubes is warranted. The staff will use the vendor's analyses and its own confirmatory analyses of these events to examine the response of the AP600 to an multiple SGTR. The staff will subsequently recommend the extent to which multiple SGTRs should be included within the design basis of the AP600.

In letters of August 21, 1992, and September 17, 1992, EPRI and Westinghouse representatives, respectively, provided comments concerning the staff's position on multiple SGTRs. Westinghouse and EPRI believe that the design basis for the AP600 should remain a single tube rupture, and that multiple tube ruptures should be analyzed on a safety margin basis, using best-estimate techniques. Westinghouse and EPRI also indicated that they expect analyses of multiple SGTRs involving up to five tubes to show that the AP600 can respond to these events without actuating the automatic depressurization system.

The staff does not share the views of Westinghouse and EPRI representatives. At this time, there is no experimental data on the performance of the AP600's passive safety systems that can be used to validate any models used for multiple SGTR analyses. The staff therefore believes that there is substantial uncertainty in any such analyses. If the ADS were to actuate in the course of an multiple SGTR, the subsequent interactions could have a serious impact on the ability of the passive systems to successfully terminate the accident. The staff expects that experimental data from planned vendor testing programs will provide an adequate basis for evaluating the response of the AP600 to an multiple SGTR.

The staff therefore maintains its stated position requiring analyses of multiple SGTRs up to five tubes for the AP600 and providing the results of these analyses in the application for design certification. The staff will determine the appropriate number of SGTRs that should be included in the AP600 design basis after evaluation of the submitted analyses.

The staff recommends that the Commission approve the position to require that analysis of multiple SGTRs involving two to five SG tubes be included in the application for design certification for passive PWRs. The staff will evaluate these analyses during the final design approval and design certification review process to help determine the number of SG tube ruptures that will be incorporated into the passive PWR design basis.

Containment Bypass Potential Resulting From SGTRs

The staff has identified an additional containment performance issue that has not adequately been addressed. Specifically, a rupture of one or more SG tubes could lead to a bypass of the containment. During a SGTR event, the SG safety or relief valves could be actuated, discharging primary system radioactive inventory outside the containment. The staff concludes that the applicant for design certification should consider providing means to mitigate this containment challenge. This issue applies to both evolutionary and passive PWR designs.

In its discussion on containment performance (Item I.J) in this Commission paper, the staff emphasized the importance of maintaining containment integrity following a postulated severe accident. In its SRM on SECY-90-016, the Commission endorsed the staff's goal of reducing the probability for conditional containment failure through the use of quantitative guidelines or alternative deterministic objectives. The EPRI requirements document states that PWR containments should be designed to provide a leak-tight barrier to prevent uncontrolled release of radioactivity in the event of a postulated (design-basis) accident. Containment bypass due to SGTRs could violate containment integrity and hamper attainment of the severe accident goals discussed in SECY-90-016.

The staff concludes that containment bypass resulting from SGTRs can be a significant challenge to containment integrity. Therefore, the staff concludes that the plant designer should consider design features that would reduce or eliminate containment bypass leakage in such a scenario. The following features could mitigate the releases associated with a tube rupture:

- a highly reliable (closed loop) steam generator shell-side heat removal system that relies on natural circulation and stored water sources;

- a system which returns some of the discharge from the steam generator relief valve back to the primary containment; or,

- increased pressure capacity on the steam generator shell side with a corresponding increase in the safety valve setpoints.

In its letter dated September 16, 1992, the ACRS indicated that it agrees with the staff's recommendations to require that the applicant for design certification of a passive or evolutionary PWR assess design features necessary to mitigate the amount of containment bypass leakage that could result from multiple SGTRs.

In a letter dated August 21, 1992, EPRI indicated that the requirements document requires that future PWRs have substantially improved capability relative to SGTRs, to minimize the potential for containment bypass and avoid repetition of past incidents, where SGTR resulted in continued lifting of steam safety valves. As such, the plant design shall be such that the complete and sudden rupture of one steam generator tube will not result in actuation of steam side safety valves. EPRI indicated that this policy issue relative to containment bypass is a design-specific issue, and relates to how a particular designer addresses the functional SGTR requirements already provided in the requirements document.

In a letter dated September 17, 1992, Westinghouse indicated that the ALWR program developed in response to this issue includes Westinghouse input. The ALWR response to mitigate this issue includes the following elements: (1) both the evolutionary and passive PWR plants address SGTR containment bypass by features which significantly reduce the potential for core damage and for release directly to the atmosphere in SGTR accident sequences; (2) ALWR utilities are seeking design features in ALWR plants which could simplify operator response to SGTR; (3) evolutionary plants are designed to terminate SGTR by operator actions with a 30 minute grace period; and (4) passive plants have the same operator assisted capability but also include capability to mitigate SGTR without operator action.

<u>The staff recommends that the Commission approve the position to require that the applicant for design certification for a passive or evolutionary PWR assess design features to mitigate the amount of containment bypass leakage that could result from SG tube ruptures.</u> The applicant or plant designer should consider the mitigation features that would likely be available following a postulated severe accident. The staff concludes that PWR designers should assess such features and address the desirability of this mitigation function. The staff will review this issue when it performs the design certification review.

S. PRA Beyond Design Certification

A plant-specific PRA is an excellent method for assessing overall plant safety and integrating plant systems and human interactions. Careful review of a PRA can also reveal important engineering evaluations, assumptions, and uncertainties. In the design and design review processes, PRA insights can be used to select among design options, strengthen the design against previously known vulnerabilities, characterize the design, and evaluate the balance in the design between severe accident prevention and mitigation.

At the COL stage, the COL applicant will be able to provide site-specific information and detailed design information that was not available during the certification process. The COL applicant will be required to update the design-specific PRA to reflect site-specific information before COL issuance. During the construction stage, the COL holder will also be able to consider as-built information. Experience has shown that subtle design interfaces involving support systems, systems interactions, or man-machine interfaces can significantly affect the risk profile of a plant.

The staff concludes that updated PRA insights, if properly evaluated and utilized, can strengthen programs and activities in areas such as training, emergency operating procedure development, reliability assurance, maintenance, and 10 CFR 50.59 evaluations. Therefore, the PRA should be revised to account for site-specific information, first-of-a-kind engineering developments, as-built (plant-specific) information refinements in the level of design detail, plant operational experience, and design changes. The COL applicant or COL holder should update the PRA to ensure that new information or design changes do not introduce new vulnerabilities or diminish the overall capability of the design to prevent and mitigate severe accidents. As plant experience data accumulates, failure rates (taken from generic data bases) and human errors assumed in the design PRA should be updated and incorporated as appropriate, into Operational Reliability Assurance Programs.

EPRI, Westinghouse, and NUMARC agree with the staff that a design-specific PRA has value and benefit, but they believe that the legal status of the PRA must be established under 10 CFR Part 52. Industry representatives expressed their desire to establish a common understanding of the legal and regulatory implications regarding the maintenance of the PRA. NUMARC, for example, stated that an ALWR design-specific PRA contains no unique or original design information that is not already reflected in associated SSAR Chapters. Accordingly, NUMARC believes that a design-specific PRA should be in neither Tier 1 nor Tier 2, but rather should be used as an analytical tool to assist the applicant and the NRC staff in evaluating the safety of the plant design.

In its letter of September 16, 1992, ACRS agreed that it is worthwhile for a plant operator to have an updated PRA. However, the ACRS was concerned about how the staff intends to use the updated PRA, how the staff thinks the licensee should use the updated PRA, and what should be required to update or keep the PRA current.

The staff is considering additional regulatory requirements to address revising the design-specific PRA after it has been completed and will discuss this issue in a future Commission paper regarding the form and content of a combined license.

T. Control Room Annunciator (Alarm)[1] Reliability

The annunciator system in a nuclear power plant provides a "first alert" to the control room operator of an abnormal state in the plant, usually over the full spectrum of transients from the malfunctioning of a single piece of equipment to the development of an abnormal state of one or more critical process parameters. The annunciator system also focuses the operator's attention on the location and nature of the malfunction or disturbance. The extent to which this is achieved depends upon the design features of the annunciator system.

Recent events at operating U.S. nuclear plants involving the loss of the plant annunciator system have revealed that the power supplies of these systems are vulnerable to single failures. At present, the NRC has no requirements specific to the annunciator system. The acceptance criteria and guidelines for I&C systems (Appendix A to SRP, Section 7.1) developed from the GDC for the I&C, control room, and protection and reactivity control systems, do not specifically include the annunciator system. IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," states that protection systems design should provide the operator with information pertinent to its own status and to the generating station's safety. In a few special cases, specific alarms are required to comply with regulatory requirements because they are essential for the manual initiation of protective actions.

When the operating U.S. plants were being designed, the international community generally observed the same requirements as those in the U.S. One of the few exceptions is International Electrotechnical Commission Publication 231A, Supplement to Publication 231, "General Principles of Nuclear Reactor Instrumentation," 1969. This publication gives specific requirements for the design of safety alarms but does not list their functional requirements.

The international requirements changed in 1984 when the International Atomic Energy Agency published Safety Guide D8, "Safety-Related Instrumentation and Control Systems for Nuclear Power Plants." This Safety Guide discusses top-level requirements for those I&C systems, including the control room annunciator system, that perform functions important to safety but are not part of the traditional safety systems.

Safety Guide D8 recommends a method for determining the relative importance to safety and the general principles for developing graded requirements for design features that determine the reliability and availability of these I&C

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[1] For the purposes of this paper, the annunciator system is considered to consist of sets of alarms (which may be displayed on tiles, video display units (VDUs), or other devices) and sound equipment; logic and processing support; and functions to enable operators to silence, acknowledge, reset and test alarms.

systems. The staff discussed the need for such classification in some detail in SECY-91-292, "Digital Computer Systems for Advanced Light-Water Reactors," dated September 16, 1991.

The EPRI URDs for both evolutionary and passive ALWR plants states the following requirements:

> The main control room (MCR) shall contain compact, redundant operator workstations with multiple display and control devices that provide organized, hierarchical access to alarms, displays, and controls. Each workstation shall have the full capability to perform MCR functions as well as support division of tasks between two operators.
>
> The display and control features shall be designed to satisfy existing regulations, for example: separation and independence requirements for Class 1E circuits (IEEE Standard 384); criteria for protection systems (IEEE Standard 279); and requirements for manual initiation of protective actions at the systems level (Regulatory Guide 1.62). The M-MIS designer shall use existing defensive measures (e.g., segmentation, fault tolerance, signal validation, self-testing, error checking, and supervisory watchdog programs), as appropriate, to ensure that alarm, display, and control functions provided by the redundant workstations meet these standards.

Thus, EPRI requires compact workstations with full capability to perform control room functions with fully organized alarms, displays, and controls. These workstations, including the alarms, are to be redundant and meet the requirements for independence and separation of Class 1E and associated circuits described in IEEE 384, "Criteria for Independence of Class 1E Equipment and Circuits." This means that independence and separation must be provided between Class 1E and non-Class 1E circuits, even though the alarm, display, and control devices are to be located in a workstation. The alarm system is considered nonsafety-related, and, therefore, the nonsafety-grade alarm circuits must be separated from interfacing Class 1E circuits. The requirements for redundancy also apply to the power supplies associated with these work stations. These requirements form a set of graded requirements for the alarm, control, and indication functions that implement the classification approach discussed in SECY-91-292.

The staff concludes that additional requirements for ALWR alarm systems are necessary to minimize the problems experienced by operating nuclear power plants, such as the total loss of annunciators because of problems with their power supplies. The EPRI requirements for redundant workstations and displays that include the alarm functions are adequate for these stations.

<u>The staff recommends that the Commission approve the position that the alarm system for ALWRs should meet the applicable EPRI requirements, as discussed above, for redundancy, independence, and separation. In addition, alarms that</u>

<u>are provided for manually controlled actions for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions, shall meet the applicable requirements for Class 1E equipment and circuits.</u>

III. Issues Limited to Passive Designs

A. Regulatory Treatment of Nonsafety Systems in Passive Designs

In contrast to both the current generation of LWR and the evolutionary ALWRs, the passive ALWR designs rely on safety systems that use the driving forces of buoyancy, gravity, and stored energy sources. These passive systems supply safety-injection water, provide core and containment cooling, and perform other functions. There are no pumps in these passive safety systems, and all valves are powered by dc electric power from batteries, are air-operated, or use check valves actuated by the pressure differential across the valve. EPRI and the passive reactor vendors contend that these designs do not include safety-grade ac electric power.

The passive ALWR designs also include nonsafety-grade active systems to provide defense-in-depth capabilities for reactor coolant makeup and decay heat removal. These systems serve as the first line of defense in the event of transients or plant upsets to reduce challenges to the passive systems. These active systems include: (1) the chemical and volume control system and control rod drive system, which provide reactor coolant makeup for the AP600 and SBWR, respectively; (2) the reactor shutdown cooling system and backup feedwater system for PWR decay heat removal, and the reactor water cleanup system for BWR decay heat removal; (3) the fuel pool cooling and cleanup system for spent fuel decay heat removal; and (4) the associated systems and structures that are needed to support these functions, including nonsafety standby diesel generators. In addition, the passive ALWR designs include nonsafety-grade active systems (such as the control room HVAC system) for mitigation of the radiological consequences of an accident. Many of these systems traditionally have been safety-grade systems, but in the passive plants, they are not designed to meet safety-grade criteria, and credit is not taken for them in the Chapter 15 licensing design-basis accident analyses. In SECY-90-406, "Quarterly Report on Emerging Technical Concerns," dated December 17, 1990, the staff identified the role of these nonsafety systems in the passive designs as an emerging technical issue.

Associated with the new, passive design approach, the licensing design-basis analysis relies solely on the passive safety systems to demonstrate compliance with the acceptance criteria for various design-basis transients and accidents. However, uncertainties remain concerning the performance of the unique passive features and overall performance of core and containment heat removal because of a lack of a proven operational performance history. For example, there are uncertainties about the performance of check valves in the passive safety systems, which operate at low differential pressures provided by natural circulation or gravity injection. These low pressures may not provide sufficient force to fully open sticking check valves (that is, pumped ECCSs are more likely to overcome stuck valves). These uncertainties enhance the importance of the active nonsafety systems in providing the defense-in-depth to prevent and mitigate accidents and core damage. Therefore, the staff's review of the passive designs requires an evaluation of not only the passive safety systems, but also the functional capability and availability of the active nonsafety systems to provide significant defense-in-depth and accident and core damage prevention capability.

For active systems that perform defense-in-depth functions, the EPRI requirements document for passive designs specifies requirements concerning performance and systems and equipment design. These include radiation shielding requirements (to permit access following an accident), redundancy, availability of nonsafety-grade electric power, and protection against internal hazards. The requirements also address safety analysis and testing to demonstrate system capability to satisfy defense-in-depth considerations. EPRI does not currently provide specific requirements for the reliability of these systems. However, in response to staff questions, EPRI has indicated that it is evaluating specific reliability targets and other measures to provide confidence that the passive plants will meet performance requirements. These requirements will address both passive safety and active nonsafety systems.

In addition, technical specification development is a subset of the overall regulatory treatment of the passive designs. The staff is evaluating the need to establish reliability-based technical specifications for passive designs. This evaluation will determine which systems and components (including certain nonsafety systems) require the imposition of technical specifications, and the parameters of the technical specifications. The Reliability Assurance Program is expected to strongly influence these technical specifications.

Since the passive ALWR design philosophy departs from current licensing practices, new regulatory and review guidance is necessary so that the staff can appropriately review the AP600 and SBWR submittals. Significant decisions need to be made concerning the scope of staff review of the nonsafety systems and reliance on the passive systems. The staff will not require that the active systems meet all the safety-grade criteria, but there should be a high level of confidence that risk-significant active systems are designed in accordance with their performance/reliability missions to ensure their availability when needed.

The staff has held several meetings with EPRI to determine steps needed to resolve the issue of regulatory treatment of active nonsafety systems, and define the scope of requirements and acceptance criteria to ensure that they have adequate capability and availability when required. In a meeting between NRC and the Utility Steering Committee on January 22, 1993, an agreement was reached for an overall process for determining the regulatory treatment of nonsafety systems, and importance of passive systems and/or components for meeting NRC Safety Goals and requirements. On February 23, 1993, EPRI submitted a document containing a draft proposed process. This document is under staff review.

The staff is still evaluating this issue for the passive plant designs. The discussion provided in this section is to inform the Commission of the current status of the issue. The staff's proposed resolution of this issue will be provided to the Commission in a separate Commission paper, which will discuss the regulatory treatment of nonsafety systems in passive designs.

B. Definition of Passive Failure

Appendix A to 10 CFR Part 50 states that any applicant must design against single failure of passive components in fluid systems important to safety, where a single failure is defined as an occurrence which results in the loss of a component's capability to perform its intended safety functions. Fluid and electric systems are considered to be designed against an assumed single failure if the system maintains its ability to perform its safety functions in the event of a single failure of either any active component (assuming passive components function properly) or a passive component (assuming active components function properly). However, the introduction to Appendix A to 10 CFR Part 50 notes that the conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

SECY-77-439, "Single Failure Criterion," describes how the staff was using the single failure criteria in its reactor safety review process. As discussed in that paper, an active failure in a fluid system means (1) the failure of a component which relies on mechanical movement to complete its intended function on demand, or (2) an unintended movement of the component. Examples include the failure of a motor- or air-operated valve to move or to assume its correct position on demand, spurious opening or closing of a motor- or air-operated valve, or the failure of a pump to start or stop on demand. In some instances, such failures can be induced by operator error.

A passive failure in a fluid system means a breach in the fluid pressure boundary or a mechanical failure which adversely affects a flow path. Examples include the failure of a simple check valve to move to its correct position when required, the leakage of fluid from failed components (such as pipes and valves) particularly through a failed seal at a valve or pump, or line blockage. Motor-operated valves which have the source of power locked out are allowed to be treated as passive components.

In past licensing reviews, the staff has been inconsistent in its treatment of passive failures in fluid systems. Specifically, the staff imposed a passive failure in addition to the initiating event but not in others. The staff has determined that, in most instances, the probability of most types of passive failures in fluid systems is sufficiently small that they need not be assumed in addition to the initiating failure in application of the single failure criterion to ensure the safety of a nuclear power plant.

In particular, staff practice has normally been to treat check valves, except for containment isolation systems, as passive devices (rather than active devices) during transients or design-basis accidents. However, the staff is considering redefining check valve failure as an active failure. This change appears necessary because safety-related check valves in the passive designs will operate under different conditions (low flow and pressure without pump pressure to open valves) than current generation reactors and evolutionary designs. In addition, they have increased safety significance to the operation of the passive safety systems, and operating experience has shown that they have a lower reliability than originally anticipated. Redefining check valve failure in this manner would cause these valves to be evaluated in a more stringent manner than that used in previous licensing reviews.

The staff is still evaluating this issue for the passive plant designs. The staff's proposed resolution of this issue will be provided to the Commission in a separate Commission paper, which will discuss the regulatory treatment of active nonsafety systems in passive designs.

C. SBWR Stability

In BWRs, thermal-hydraulic instabilities can cause oscillations that can result in violation of the minimum critical power ratio (MCPR) safety limits. The staff has concluded that GE's analytical codes have been sufficiently validated to demonstrate the stability of the ABWR design. However, the codes that GE is using have not yet been adequately validated for the passive BWR design.

As discussed in SECY-91-273, "Review of the Vendor's Test Programs to Support the Design Certification of Passive Light-Water Reactors," the staff determined that an early NRC assessment is needed. This assessment should address the vendor's analytical and experimental basis for demonstrating nuclear/thermal-hydraulic stability. In addition, it should identify any tests or analyses that may be needed to support the staff's technical evaluations of the issue. The NRC staff and its consultant, Oak Ridge National Laboratories (ORNL), have reviewed the thermal-hydraulic stability characteristics of the SBWR based on preliminary design information provided by GE. This assessment included calculations with the LAPUR computer code developed by NRC and ORNL. These calculations showed that, while the system appears to be very stable under normal operating conditions, certain abnormal operating conditions might be reached under credible transient sequences. These abnormal conditions can result in the onset of density-wave power and flow oscillations. In addition, a low-flow and low-power instability caused by a "geysering" effect between parallel channels has been identified as a concern during normal operating transients such as start-up and shutdown.

On December 6, 1991, the staff met with EPRI and GE to discuss the EPRI/GE response to the staff's conclusions. Specifically, the staff concluded that more extensive SBWR stability studies are needed and that codes which have been validated against thermal-hydraulic tests representative of the SBWR design (including the large open chimney) would be needed to perform these studies. EPRI and GE informed the staff that the chimney design has been changed and that existing experiments are representative of the divided chimney now employed. GE also indicated it will validate its codes for density-wave instability studies against these experiments and will provide the results of this work for NRC review. The geysering instability is being studied using small-scale experiments performed by a Japanese partner to GE. The Japanese SAFAR code will be validated against these experiments and used for analytical prediction of stable operating boundaries. GE plans to recommend start-up/shutdown procedures similar to those used in the Dutch Dodewaard reactor to avoid geysering instability. In addition, EPRI and GE believe that the SBWR is not vulnerable to a loop-type instability reported by the Japanese; rather, EPRI and GE contend that this instability was characteristic of the experimental apparatus used.

In SECY-92-339, "Evaluation of the General Electric Company's (GE's) Test Program to Support Design Certification for the Simplified Boiling Water Reactor (SBWR)," the staff noted that GE has modified the SBWR conceptual design. The staff also noted that GE has identified existing experimental data, which they believe constitutes appropriate validation of codes to be used for stability studies. EPRI and GE have indicated agreement with the staff that such studies will be needed to confirm the stability of the SBWR during various transient scenarios (including ATWS). However, GE has not provided sufficient information to permit NRC evaluation of the applicability and sufficiency of the foreign experiments they have identified for use during code validation. The vendor has agreed to make this information available to the NRC as soon as it obtains permission from the foreign sources. Until these experiments can be reviewed by NRC, the potential need for additional experiments to support stability evaluations for design certification remains open.

The staff considers this a technical issue and expects to resolve this issue with GE through its normal review of the SBWR design certification application and through its review of GE's SBWR testing program. The staff will interact with the Commission if additional policy issues are identified during its review of the SBWR application or vendor SBWR testing program.

D. Safe Shutdown Requirements

GDC 34 of Appendix A to 10 CFR Part 50 requires that a residual heat removal system be provided to remove residual heat from the reactor core so that specified acceptable fuel design limits (SAFDLs) are not exceeded. RG 1.139 and Branch Technical Position (BTP) 5-1 implement this requirement and set forth conditions for cold shutdown 93.3 °C (200 °F) for a PWR and 100 °C (212 °F) for a BWR) using only safety-grade systems within 36 hours. The RG presents the basis for this requirement, as follows:

> ...even though it may generally be considered safe to maintain a reactor in a hot standby condition for a long time, experience shows that there have been events that required eventual cooldown and long-term cooling until the reactor coolant system was cold enough to perform inspection and repairs. It is therefore obvious that the ability to transfer heat from the reactor to the environment after a shutdown is an important safety function for

> both PWRs and BWRs. Consequently, it is essential that a power plant have the capability to go from hot-standby to cold-shutdown conditions...under any accident conditions.

Because passive ALWR designs use passive heat removal systems for decay heat removal, they are limited by the inherent ability of the passive heat removal processes. These designs cannot reduce the temperature of the reactor coolant system below the boiling point of water for the heat to be transferred to the in-containment refueling water storage tank of the AP600 or the isolation condenser of the SBWR. Even though active shutdown cooling systems are available to bring the reactor to cold-shutdown or refueling conditions, these active RHR systems are not safety-grade and do not comply with the guidance of RG 1.139 or BTP 5-1.

EPRI states that it is not necessary for passive safety systems to be capable of achieving cold shutdown. EPRI bases this contention on the belief that the passive decay heat removal (DHR) systems have an inherently high long-term reliability. The EPRI requirements document for passive plant designs states that the passive ALWR designs will employ a redundant safety system for both the hot-standby and long-term cooling modes. In addition, it defines safe shutdown as 215.6 °C (420 °F). EPRI has indicated that it meets GDC 34 requirements because redundant passive decay heat removal systems can operate at full RCS pressure and place the reactor in the long-term cooling mode immediately after shutdown. Additionally, EPRI requires that operation of the plant in the long-term cooling mode be automatic, eliminating operator actions to cool down the plant. Also, operation of the passive DHR system does not require any ac power or pumps. EPRI further states that the nonsafety systems that will take the plant to cold-shutdown conditions "...are highly reliable in their own right...and a failure in these systems would not prevent the plant from achieving cold shutdown."

The staff is currently evaluating the EPRI position with respect to this issue to assess the acceptability of their proposed alternative approach for meeting GDC 34. The long-term DHR capability of the proposed passive systems offers potential advantages over current active systems. However, the staff must resolve several issues before reaching a final position on this matter. These issues include reliability criteria for the nonsafety systems which have the capability to bring the plant to cold shutdown and the acceptability of 215.6 °C (420 °F) as a safe, long-term state. The staff's proposed resolution of this issue will be provided to the Commission in a separate Commission paper, which will discuss the regulatory treatment of nonsafety systems in passive designs.

E. Control Room Habitability

GDC 19 of Appendix A to 10 CFR Part 50 states that adequate radiation protection shall be provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. In current plants, safety-grade filtered control room HVAC systems with charcoal absorbers are used to ensure that radiation doses to operators could be maintained within the GDC 19 criteria in the event of an accident.

In SRP Section 6.4, the staff defined the acceptable operator dose criterion in terms of specific whole body and organ doses (5 rem to the whole body, and 30 rem each to the thyroid and skin). Recently, the NRC embraced the principal recommendations of Publication No. 26 of the International Commission on Radiological Protection in the promulgation of a major revision of 10 CFR Part 20. The adoption of these recommendations, which include use of the effective dose equivalent, did not change the dose criteria for control room operators to conform with GDC 19.

Originally, EPRI proposed an exposure limit for control room operators of 5 rem whole body, 75 rem skin, and 300 rem thyroid. EPRI stated that each operator would be provided with individual breathing apparatus and protective clothing, if required, to meet regulatory limits. The staff determined that EPRI's requirements for the thyroid and beta skin doses were not adequately justified. The staff indicated to EPRI that the long-term use of a breathing apparatus during design-basis accidents has never been allowed. More importantly, the long-term use of a breathing apparatus is likely to degrade operator performance during and following an accident.

EPRI also stated that the control room would be designed to be maintained during a 72-hour period as the primary location from which personnel can safely operate in the event of an accident. It was and is the staff's position that, depending upon the accident, the required duration may be much longer than 72 hours. GDC 19 states that "adequate radiation protection shall be provided to permit access to and occupancy of the control room under accident conditions...for the duration of the accident." Consequently, the staff concluded that analyses of control room habitability should consider the duration of the accident (which may extend well beyond the EPRI-proposed 72-hour period) as the design basis.

In order to resolve this matter, the staff proposed that EPRI and the vendors provide a high level of assurance that the control room ventilation system will be available when needed. Because the system may not need to meet all of the safety-grade criteria, it may be appropriate to allow some credit for nonsafety-grade ventilation and filtration systems based on reliability considerations. The extent of this credit will be determined as part of the staff's review of the regulatory treatment of nonsafety systems. It should be noted that unlike the case of core cooling, there is no passive safety-grade system for defense-in-depth of control room habitability.

In its letter of May 5, 1992, EPRI proposed an alternative which would use a safety-grade pressurization system capable of being recharged remotely after 72 hours. In that enclosure, EPRI stated that Volume III of the Utility Requirements Document would be revised to require: (1) a passive, safety-grade control room pressurization system which would use bottled air to keep operator doses within the limits of GDC 19 and SRP Section 6.4, Revision 2 for the first 72 hours of the event, and (2) safety-grade connections for the pressurization system to allow use of offsite, portable air supplies if needed after 72 hours to minimize operator doses. The staff agrees with EPRI's commitment to limit the operator doses to those specified in GDC 19 and SRP Section 6.4, Revision 2. However, the staff is still evaluating the proposal to utilize a safety-grade pressurization system and has serious reservations concerning the feasibility and the capability of a pressurization system to maintain the control room habitability.

In its letter of August 17, 1992, ACRS indicated that they had discussed the subject of control room habitability with EPRI and the staff during a meeting on June 4 and 5, 1992. At that meeting, the staff told ACRS that they were evaluating the EPRI proposal utilizing the safety-grade pressurization system. ACRS indicated that they had several comments regarding the design features of the passive control room pressurization system proposed by EPRI. ACRS stated that the staff should take these comments into account in performing its evaluation, and that ACRS may provide additional recommendations after the staff has completed its evaluation.

The staff is currently reviewing the new severe accident source term proposed by EPRI in conjunction with the staff's technical update of the TID-14844 source term. The estimated potential radiological consequences to the passive plant control room operators during a severe accident will depend on the outcome of the forthcoming resolution of severe accident source term. Specifically, this outcome will include chemical forms of fission products, release fractions, and release timing. In addition, the control room habitability assessment is further dependent upon the fission product removal processes inside, as well as outside, of the primary containment before it reaches the control room air intake and the control building that houses the control room. Therefore, the staff is unable to complete its control room habitability assessment until issues concerning the source term and its behavior mechanism are satisfactorily resolved.

The staff plans to present its proposed resolution of this issue in a separate Commission paper, which will discuss issues related to source term.

F. Radionuclide Attenuation

EPRI and the passive ALWR designers rely on assumptions involving fission product removal inside containment by natural removal effects and holdup by the secondary building and piping systems. A containment spray system is not mandated by the EPRI requirements document for passive plant designs. The staff is concerned about the uncertainty in quantifying the holdup phenomena in the auxiliary building and that use of the auxiliary building for holdup may require imposition of additional restrictions on the auxiliary building during normal operation, with which the licensee may have difficulty complying.

This issue also affects the control room habitability issue discussed in Section III.E of this report. The relationship arises because the industry indicates that fission products will be removed before they reach the control room air intake or the control building that houses the control room.

The staff is still evaluating this issue as well as the need for a containment spray system for the passive plant designs. The staff is also evaluating whether credit for fission product attenuation in the main steamlines and condenser is appropriate for the passive BWR design. This question arises because the main steamlines downstream of the main steam isolation valves and associated condenser are not designed to withstand the SSE, as defined in Section III.c of 10 CFR Part 100. The staff concludes that plateout of radioactive iodine on the main steam pipe and condenser surfaces following a severe accident can realistically provide significant dose mitigation. Several technical references indicate that particulate and elemental iodines would be expected to deposit on surfaces with deposition rates varying with temperature, pressure, gas composition, surface material, and particulate size.

The staff's proposed resolution of this issue will be presented in a separate Commission paper, which will discuss source term related issues.

G. Simplification of Offsite Emergency Planning

EPRI has proposed to significantly simplify offsite emergency planning for passive designs because of EPRI's estimated low probability of core damage and, in the event of a core damage accident, the assurance of containment integrity and low offsite dose. EPRI's proposal would eliminate requirements for early notification of the public, detailed evacuation planning, and provisions for exercising the offsite plan. The onsite emergency plan and limited offsite actions would be retained. EPRI has identified this matter as a plant optimization subject.

During a meeting with the staff on January 30, 1992, EPRI proposed to work with the staff to define a process for addressing simplification of emergency planning. This simplification would include developing technical criteria and methods that, if met, would justify such action. It would also include defining the process for implementing this approach. The results of this effort would be used as input to a generic rulemaking proposal to be initiated by NUMARC.

The staff concludes that certain modifications to the emergency planning requirements of 10 CFR Part 50 and the siting criteria in 10 CFR Part 100 may be appropriate for the passive designs based on their unique characteristics. However, an agency determination on these issues will require evaluation of detailed design information. The staff concludes that the unique characteristics of these designs should be taken into account in determining the extent of emergency planning required for the plume exposure pathway emergency planning zone. Any decision on emergency planning requirements for the passive design should reflect a plant's ability to prevent the significant release of radioactive material or to provide very long delay times before a release for all but the most unlikely events. Before relaxing emergency planning requirements, the staff will require a high degree of assurance that all potential containment bypass accident sequences have a very low likelihood. The lack of information concerning source term and risk precludes further staff evaluation of the emergency preparedness requirements for the passive reactor designs at this time. Moreover, the issue is complicated by the fact that the promulgation of emergency planning requirements following the TMI-2 accident was not premised on any specific assumptions about severe accident probability. Hence, as a policy matter, it may be that even very low calculated probability values should not be considered a sufficient basis for changes to emergency planning requirements.

The staff will evaluate this issue for the passive plant designs when sufficient supporting information becomes available. The staff plans to update the status of this review in a separate Commission paper.

H. Role of the Passive Plant Control Room Operator

In SECY-91-272, "Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants," dated August 27, 1991, the staff discussed the role of the operator in a passive plant control room. Specifically, operators in a passive plant control room may use nonsafety-related systems and active "investment protection" systems as the primary means to mitigate transients and accidents. Operators will use these systems, before safety-related passive systems are initiated, when responding to transients and accidents. The design of safety-related systems in the passive plant differs significantly from the design of safety-related systems in current operating plants and in evolutionary plant designs.

To safely operate a passive plant, the operator must understand the operation of the "investment protection" systems and their interfaces with the safety-related passive systems. Passive plant operators will be required to perform new functions and tasks unlike those for evolutionary plants. These new functions and tasks will be associated with the new operational philosophy noted above, the increase in automation, and the greater use of advanced technology in passive plant designs. These new functions and tasks will likely involve greater reliance on monitoring and decision-making rather than performing actions directed in procedures. Thus, the design process must carefully define the operator's role to ensure that it properly develops the man/machine interface design to facilitate these functions and tasks.

EPRI stated that the ALWR Program has provided for "man-in-the-loop" testing during first-time engineering, as specified within Chapter 10 of the EPRI URD. EPRI also requires a full scope control room design simulator for this testing. To correct the problem of insufficient focus on the operator in previous designs, EPRI indicated that this requirement should adequately ensure that the human component in the man/machine interface system is explicitly included. However, EPRI maintains that the difference in the role of the operator in a passive plant control room is limited to the details and timing of the actions performed. The staff will consider the operator's role in its review of each passive plant applicant's control room design under 10 CFR Part 52.

The staff concludes that an extensive man-in-the-loop test and evaluation program will be necessary for the passive plant control room designs. This testing will address the extent of differences in the operator's role in a passive plant control room since it will simulate tasks necessary for maintaining plant safety following an event. Such testing would likely require a fully functional integrated control room prototype to demonstrate that the passive designs properly consider the operator's role for ensuring plant safety.

In its letter of August 21, 1992, EPRI restated that the approach to operating passive designs is the same as for evolutionary or existing designs. The staff does not agree with this position. However, the EPRI ALWR Program has provided for man-in-the-loop testing and a full scope control room design simulator to ensure that the human component is explicitly considered and is acceptable. The staff has continued discussions with EPRI and passive plant vendors and believes this approach will resolve the differences in the position on operating philosophy.

In its letter of September 16, 1992, ACRS agreed with the staff that sufficient man-in-the-loop testing and evaluation should demonstrate that the operator's functions and tasks are properly integrated into the man/machine interface design.

Therefore, the staff recommends that the Commission approve the position that sufficient man-in-the-loop testing and evaluation must be performed. In addition, a fully functional integrated control room prototype is likely to be necessary for passive plant control room designs to demonstrate that functions and tasks are properly integrated into the man/machine interface design.

These requirements will be incorporated into the DAC. Each applicant may provide justification that a control room prototype of reduced scope is sufficient to ensure that functions and tasks are properly integrated in the man/machine interface design.

ALWR ISSUES CROSS-REFERENCE MATRIX

Category	Issue	Title	Commission Papers
I. SECY-90-016 Issues	A.	Use of Physically Based Source Term	86-228 88-203 89-013 89-153 89-228 89-341 90-016 90-307 90-329 90-341 90-353 92-127 Draft-1[1]
	B.	ATWS	89-153 89-228 90-016 90-353 Draft-1
	C.	Mid-Loop Operation	89-228 90-016 90-353 Draft-1
	D.	Station Blackout	89-013 89-153 89-228 90-016 90-329 90-353 Draft-1
	E.	Fire Protection	89-013 89-228 90-016 90-353 Draft-1

[1] "Draft-1" refers to the draft Commission paper, "Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements," which was forwarded to the Commission on February 20, 1992, and made available to the public on February 27, 1992.

Category	Issue	Title	Commission Papers
I. SECY-90-016 Issues (cont.)	F.	Intersystem LOCA	89-153 89-228 90-016 90-353 Draft-1
	G.	Hydrogen Control	89-013 89-153 89-228 90-016 90-329 90-353 Draft-1
	H.	Core Debris Coolability	89-153 89-228 90-016 90-353 92-092 Draft-1
	I.	High-Pressure Core Melt Ejection	89-228 90-016 90-353 92-092 Draft-1
	J.	Containment Performance	89-228 90-016 90-353 91-273 92-092 Draft-1
	K.	Dedicated Containment Vent Penetration	89-153 89-228 90-016 90-329 90-353 92-092 Draft-1
	L.	Equipment Survivability	89-228 90-016 90-353 Draft-1
	M.	Elimination of OBE	89-013 90-016 90-329 90-353

Category	Issue	Title	Commission Papers
I. SECY-90-016 Issues (cont.)	M.	Elimination of OBE (cont.)	91-135 Draft-1 Draft-2[2]
	N.	In-Service Testing of Pumps and Valves	89-228 90-016 90-353 91-273 Draft-1
II. Other Evolutionary and Passive Design Issues	A.	Industry Codes and Standards	91-273 Draft-1
	B.	Electrical Distribution	91-078 Draft-1
	C.	Seismic Hazard Curves and Design Parameters	91-135 Draft-1
	D.	Leak-Before-Break	89-013 Draft-1
	E.	Classification of Main Steamlines in BWRs	Draft-1
	F.	Tornado Design Basis	Draft-1
	G.	Containment Bypass	Draft-1
	H.	Containment Leak Rate Testing	89-013 89-228 91-273 Draft-1
	I.	Post-Accident Sampling System	Draft-1
	J.	Level of Detail	90-241 90-377 Draft-1
	K.	Prototyping	91-074 91-273 Draft-1

[2]"Draft-2" refers to the draft Commission paper, "Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," which was forwarded to the Commission on June 25, 1992, and made available to the public on July 1, 1992.

Category	Issue	Title	Commission Papers
II. Other Evolutionary and Passive Design Issues (cont.)	L.	ITAAC	91-178 91-210 92-053 92-196 92-214 92-287 92-294 Draft-1
	M.	Reliability Assurance Program	89-013 92-133 Draft-1
	N.	Site-Specific PRAs and Analyses of External Events	89-013 Draft-1 Draft-2
	O.	SAMDAs	91-229 Draft-1
	P.	Generic Rulemaking Related to Design Certification	91-262 Draft-1
	Q.	Defense Against Common-Mode Failures in Digital I&C Control Systems	91-292 Draft-2
	R.	Multiple SG Tube Ruptures	92-133 Draft-2
	S.	PRA Beyond Design Certification	Draft-2
	T.	Control Room Annunciator Reliability	Draft-2
III. Issues Limited to Passive Design	A.	Regulatory Treatment of Active Nonsafety Systems	89-013 90-406 92-133 Draft-1 Draft-2
	B.	Definition of Passive Failure	77-439 Draft-1
	C.	Thermal-Hydraulic Stability of the SBWR	89-153 91-273 92-339 Draft-1

Category	Issue	Title	Commission Papers
III. Issues Limited to Passive Design (Cont.)	D.	Safe Shutdown Requirements	Draft-1 92-133
	E.	Control Room Habitability	92-133 Draft-1 Draft-2
	F.	Radionuclide Attenuation	92-127 92-133 Draft-1
	G.	Simplification of Offsite Emergency Planning	88-203 Draft-1
	H.	Role of the Passive Plant Control Room Operator	91-272 Draft-2

COMMISSION PAPERS APPLICABLE TO ALWRs

SECY-77-439, "Single Failure Criterion," August 17, 1977.

SECY-86-228, "Introduction of Realistic Source-Term Estimates into Licensing," August 6, 1986.

SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," May 25, 1988.

SECY-88-202, "Standardization of Advanced Reactor Designs," July 15, 1988.

SECY-88-203, "Key Licensing Issues Associated with DOE-Sponsored Advanced Reactor Designs," July 15, 1988.

SECY-89-012, "Staff Plans for Accident Management Regulatory and Research Programs," January 18, 1989.

SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light-Water Reactors (ALWRs)," January 19, 1989.

SECY-89-153, "Severe Accident Design Features of the Advanced Boiling-Water Reactor (ABWR)," May 10, 1989.

SECY-89-178, "Policy Statement Integration," June 9, 1989.

SECY-89-228, "Draft Safety Evaluation Report on Chapter 5 of the Advanced Light-Water Reactor Requirements Document," July 28, 1989.

SECY-89-341, "Updated Light-Water Reactor (LWR) Source-Term Methodology and Potential Regulatory Applications," November 6, 1989.

SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," January 12, 1990.

SECY-90-065, "Evolutionary and Passive Advanced Light Water Reactor Resources and Schedules," March 7, 1990.

SECY-90-146, "Process, Schedule, and Resources for the Review of Evolutionary and Passive Advanced Light-Water Reactors," April 20, 1990.

SECY-90-241, "Level of Detail Required for Design Certification Under Part 52," July 11, 1990.

SECY-90-307, "Impacts of Source-Term Timing on NRC Regulatory Positions," August 30, 1990.

SECY-90-313, "Status of Accident Management Program and Plans for Implementation," September 5, 1990.

SECY-90-329, "Comparison of the General Electric Advanced Boiling-Water Reactor (ABWR) Design and the Electric Power Research Institute's (EPRI's) Advanced Light-Water Reactor (ALWR) Requirements Document," September 20, 1990.

SECY-90-341, "Staff Study on Source-Term Update and Decoupling Siting from Design," October 4, 1990.

SECY-90-353, "Licensing Review Basis for the Combustion Engineering, Inc. System 80+ Evolutionary Light-Water Reactor," October 12, 1990.

SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," November 8, 1990.

SECY-90-406, "Quarterly Report on Emerging Technical Concerns," December 17, 1990.

SECY-91-074, "Prototype Decisions for Advanced Reactor Designs," March 19, 1991.

SECY-91-078, "Chapter 11 of the Electric Power Research Institute's (EPRI's) Requirements Document and Additional Evolutionary Light-Water Reactor (LWR) Certification Issues," March 25, 1991.

SECY-91-135, "Conclusions of the Probabilistic Seismic Hazard Studies Conducted for Nuclear Power Plants in the Eastern United States," May 14, 1991.

SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions," May 31, 1991.

SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses," June 12, 1991.

SECY-91-210, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) Requirements for Design Review and Issuance of a Final Design Approval," July 16, 1991.

SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," July 31, 1991.

SECY-91-239, "Preapplication Reviews of Advanced LWR Designs," August 5, 1991.

SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light-Water Reactor (LWR) Designs," August 16, 1991.

SECY-91-272, "Role of Personnel and Advanced Control Rooms in Future Nuclear Power Plants," August 27, 1991.

SECY-91-273, "Review of Vendors' Test Programs to Support the Design Certification of Passive Light Water Reactors," August 27, 1991.

SECY-91-292, "Digital Computer Systems for Advanced Light-Water Reactors," September 16, 1991.

SECY-91-348, "Issuance of Final Revision to Appendix J to 10 CFR 50, and Related Final Regulatory Guide 1.XXX (MS 021-5)," October 25, 1991.

SECY-92-030, "Integral System Testing Requirements for Westinghouse's AP600 Plant," January 27, 1992.

SECY-92-037, "Need for NRC-Sponsored Confirmatory Integral System Testing of the Westinghouse AP600 Design," January 31, 1992.

SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," February 19, 1992.

SECY-92-092, "The Containment Performance Goal, External Event Sequences, and the Definition of Containment Failure for Advanced Light-Water Reactors," March 17, 1992.

SECY-92-120, "NRC Staff Review Schedules for the Westinghouse AP600 and the General Electric (GE) Simplified Boiling-Water Reactor (SBWR) Designs," April 7, 1992.

SECY-92-127, "Revised Accident Source Terms for Light-Water Nuclear Power Plants," April 10, 1992.

SECY-92-133, "Draft Safety Evaluation Report for Volume I and Volume III of the Electric Power Research Institute's Advanced Light-Water Reactor Requirements Document," April 14, 1992.

SECY-92-134, "NRC Construction Inspection Program for Evolutionary and Advanced Reactors Under 10 CFR Part 52," April 15, 1992.

SECY-92-170, "Rulemaking Procedures for Design Certification," May 8, 1992.

SECY-92-196, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling-Water Reactor (ABWR)," May 28, 1992.

"Issues Pertaining to Evolutionary and Passive Light-Water Reactors and Their Relationship to Current Regulatory Requirements," draft Commission paper forwarded to the Commission on February 20, 1992, and made available to the public on February 27, 1992.

SECY-92-211, "NRC Confirmatory Integral System Testing for the General Electric SBWR Design," June 5, 1992.

SECY-92-214, "Development of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications," June 11, 1992.

SECY-92-219, "NRC-Sponsored Confirmatory Testing of the Westinghouse AP600 Design," June 16, 1992.

"Design Certification and Licensing Policy Issues Pertaining to Passive and Evolutionary Advanced Light-Water Reactor Designs," draft Commission paper forwarded to the Commission on June 25, 1992, and made available to the public on July 1, 1992.

SECY-92-277, "Final Safety Evaluation Report for Volume II of the Electric Power Research Institute's Advanced Light Water Reactor Requirements Document," August 10, 1992.

SECY-92-287, "Form and Content for a Design Certification Rule," August 18, 1992.

SECY-92-292, "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future LWRs," August 21, 1992.

SECY-92-294, "Acceptance Review of the Westinghouse Electric Corporation's Application for Final Design Approval and Design Certification for the AP600 Design," August 24, 1992.

SECY-92-299, "Development of Design Acceptance Criteria (DAC) for the Advanced Boiling-Water Reactor (ABWR) in the Areas of Instrumentation and Controls (I&C) and Control Room Design," August 27, 1992.

SECY-92-327, "Reviews of Inspections, Test, Analyses, and Acceptance Criteria (ITAAC) for the General Electric (GE) Advanced Boiling Water Reactor (ABWR)," September 22, 1992.

SECY-92-339, "Evaluation of the General Electric Company's (GE's) Test Program to Support Design Certification for the Simplified Boiling-Water Reactor," October 6, 1992.

SECY-92-368, "Final Rule Amending 10 CFR Part 52," October 29, 1992.

SECY-92-381, "Rulemaking Procedures for Design Certification," November 10, 1992.

SECY-92-403, "Acceptance Review of GE Nuclear Energy's (GE's) Application for Final Design Approval (FDA) and Design Certification (DC) of the Simplified Boiling-Water Reactor Design (SBWR)," December 3, 1992.