Docket File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

MAR 0 9 1994

Docket No. 52-001

Mr. John T. Larkins, Executive Director
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Mr. Larkins:

SUBJECT: REVISED PAGES FOR THE SAFETY EVALUATION REPORT ON THE ADVANCED BOILING-WATER REACTOR (ABWR) DESIGN

The advance copy of the safety evaluation report (SER) on the ABWR design was provided to the Advisory Committee on Reactor Safeguards (ACRS) in December 1993. This SER identified 14 open items remaining from the staff's review of GE Nuclear Energy's (GE's) application for design certification. The purpose of this memorandum is to provide the ACRS with the revised pages of the SER that document the basis for resolution of 13 of these issues. Accordingly, the enclosure describes the draft SSAR changes and GE actions or other information, which provide the basis for reclassifying these items.

As in the past, each of the confirmatory items will be resolved only after the staff verifies that the information provided by GE has been adequately and accurately incorporated into the ABWR standard safety analysis report (SSAR). The staff expects this information to be provided in Amendment No. 34 to GE's SSAR, and it will include several types of changes. This amendment will address the open and confirmatory items identified in the advance copy of the SER, and it will include changes resulting from (1) the staff's comprehensive review of SSAR Amendment No. 33 and the certified design material, (2) the NRR Independent Quality Review Group findings, and (3) the independent audit of the technical specifications. In addition, Amendment No. 34 will reflect SSAR changes resulting from the ACRS review and implementation of the Commission's metrication policy.

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The staff is currently updating the SER to resolve legal and editorial comments and to incorporate the enclosed revisions. The staff will also respond, in Chapter 21 of the SER, to the comments made by the ACRS in its upcoming letter. After the staff verifies that all of the required modifications have been made to the SSAR, it will request the Commission's approval to publish the "final" SER as a NUREG report.

Original Signed By:

Dennis M. Crutchfield, Associate Director
for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation

Enclosure:
Revised pages for
ABWR SER

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Dennis M. Crutchfield

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Enclosure:
Revised pages for
ABWR SER

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Enclosure

Note to recipient:

The enclosed pages are provided as markups or inserts to the advance copy of the Safety Evaluation Report related to the certification of the ABWR design, which was issued in December 1993.

Index of ABWR Open Items

Item No.	Description of Item	Advance SER Page No.
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* Corrected Open Item #

considered in evaluating the proposed design. A summary of the principal matters of the staff's review is provided in Section 1.5 of this report.

The staff provided the status of its initial review of the ABWR SSAR in a series of "draft" SERs. These draft SERs and the chapters of the SSAR that were evaluated were submitted to the Commission as follows:

Commission Paper (Date)	SSAR Chapter
SECY-91-153 (May 24, 1991)	2-6, 17
SECY-91-235 (August 2, 1991)	3, 9, 10, 11, 13
SECY-91-294 (September 18, 1991)	7
SECY-91-309 (October 1, 1991)	19
SECY-91-320 (October 15, 1991)	18
SECY-91-355 (October 31, 1991)	2, 3, 5, 6, 8, 9, 10, 12, 13, 14, 15

The staff is continuing its review and plans to issue a "final" SER as a NUREG report after GE resolves all of the open and confirmatory items identified in Sections 1.6 and 1.7 of this report, and any additional issues identified by the ACRS. The staff is presently reviewing the SSAR, including Amendment 33, to verify that the advance SER findings are still valid and supported by the SSAR. The staff is also performing a review of the SSAR, final technical specifications (TSs), and CDM to assure that this information is internally consistent. This was identified as Confirmatory Item 1.1-1 in the DFSER and is now Confirmatory Item F1.1-1. Any inconsistencies or discrepancies will be resolved before issuance of the final SER.

In a letter dated November 9, 1993, the NRC directed GE to revise the SSAR and CDM to conform with the NRC's Metrication Policy, which was published in the Federal Register (57FR46202) on October 7, 1992. The NRC requested that these revised documents be submitted prior to March 4, 1994. As stated in the November 9, 1993, letter, the staff will prepare the final SER in dual units.

In its application, GE stated that it is developing the ABWR design to meet the requirements in the Electric Power Research Institute's (EPRI) Advanced Light Water Reactor Program. The Commission requested, in a staff requirements memorandum (SRM) dated December 15, 1989, that the staff evaluate deviations that the vendor designs have with the EPRI document. This was identified as Open Item 1.1-1 in the DFSER. GE stated in a letter dated April 29, 1993, that the SSAR satisfies the objectives of the policy guidance set forth by the Commission in the above SRM. The Commission designated this response to be acceptable in COMSECY-93-040, dated August 10, 1993. This resolves DFSER Open Item 1.1-1.

This SER references information in the SSAR that GE has requested, in a letter dated November 5, 1993, to be withheld as proprietary in accordance with the provisions of 10 CFR 2.790. ~~The staff has not completed its review of GE's request for withholding.~~ This ~~is~~ WAS Open Item F1.1-1. INSERT A Several references to GE topical reports are also made in this SER. Some of these topical reports contain information that has been authorized by the Commission to be exempt

Open Item F1.1-1 *Request for withholding under § 2.790*

<u>Insert A, SER page 1-2</u>

In a letter dated March 3, 1994, GE committed to submit additional proprietary information on Chapters 11 and 18 of the SSAR and provide one or more affidavits as required by § 2.790. This is now Confirmatory Item F1.1-2.

7.2.4	Applicable regulation for control room annunciators.
7.2.6	Applicable regulation for digital instrumentation and control systems.
8.3.9	Applicable regulation for Station Blackout.
9.5.1	Applicable regulation for Fire Protection.
17.3	Applicable regulation for reliability assurance program.
19.1	Applicable regulation for seismic margins.
19.2	Applicable regulation for core debris cooling.
19.2	Applicable regulation for high pressure core melt ejection.
19.2	Applicable regulation for containment performance.
19.2	Applicable regulation for containment vent.
19.2	Applicable regulation for equipment survivability.
19.3	Applicable regulation for shutdown risk.
20.4.21	Exemption from Safety Parameter Display Console.
20.4.31	Exemption from post-accident sampling.
20.5.26	Exemption from two isolation barriers
20.5.46	Exemption from dedicated containment penetration.

1.9 Combined License Action Items

Applicants and licensees who reference the certified ABWR design will be required by the design certification rule to conform with the requirements and commitments in the DCD, which are identified in the SSAR. Many of these requirements and commitments were identified as COL action items in the DFSER. These items generally consist of programmatic, site-specific, and applicant-specific issues. An applicant will be required to describe the resolution of these items in its application for a COL. The staff requested GE to include the list of COL action items in a future amendment to Chapter 1 of the SSAR and provide an explanation in the applicable sections of the SSAR. This was identified as Open Item 1.9-1 in the DFSER.

GE incorporated the COL action items identified by the staff in the DFSER and referred to these items in the SSAR as "COL license information." The staff ~~has~~ identified some additional issues for consideration as COL action items. These included increased capability for the ACIWA system, enhanced reliability for the turbine trip logic, operating procedures for venting, testing of RCIC bypass, and materials selection. This ~~is now~~ was Open Item F1.9-1.

INSERT B-1

Open Item F1.9-1 *Additional COL Action Items*

Insert B-1, SER page 1-12

In a letter dated February 7, 1994, GE responded to the COL action items identified above and provided markups of the SSAR as appropriate. The following discussion describes the resolution of these items.

For the ACIWA system, GE identified a revision to SSAR Section 19I.3.1 and added a new Section 19.9.21 requiring that the COL applicant revise the plant-specific PRA if the ACIWA is housed in a separate building. The staff finds this response acceptable.

GE indicated that the turbine trip logic is within the ABWR design scope and provided revisions to SSAR Sections 10.2.1.2, 10.2.2.4, 10.2.4, 10.4.1.2, 10.4.1.5.1, 10.4.6.1.2, 10.4.6.2.1, 10.4.7.1, 10.4.7.2.3, 10.4.7.3, 10.4.7.5, and revised several figures in Chapter 10. This information provides more detail on the trip logic and the benefits gained in turbine availability. The staff finds these changes acceptable and determined that an additional COL action item is not needed.

GE also provided new SSAR Subsections 1A.2.5 and 1A.3.6, which added a COL action item for venting procedures. The staff finds this response acceptable.

For testing of RCIC bypass, GE modified SSAR Subsection 1A.2.23 and added Subsection 1A.3.8, which included a COL action item to perform a bypass start system test. The staff finds this response acceptable.

For materials selection, GE added a new SSAR Subsection 12.3.1.1.1, which provided design commitments that will reduce potential exposures through the use of appropriate materials. In addition, GE added SSAR Subsection 12.3.7.4, which included a COL action item. The staff finds this response acceptable.

The staff completed its review of the list of COL license information provided in the SSAR up through Amendment 33 and notified GE of missing and redundant items. GE has committed to provide an updated list in the upcoming amendment to the SSAR. One of the missing COL action items is a requirement for the COL applicant to provide an updated PRA during the construction and operation stages. The staff's recommendation for this "living" PRA is provided in the last paragraph of Section 19.1.2 of this report.

Open item F1.9-1 is now Confirmatory Item F1.9-1. Additional inserts for this item are provided for pages 12-4, 20-87, and 20-117.

at that time was

GESTAR II, Revision 10, Supplement for United States, Appendix C. However, the staff has noted deficiencies in the generic fuel licensing criteria. In its audit summary, "Audit Team Audit of GE II Fuel Design Compliance with NEDE-24011-P-A," dated March 25, 1992, the staff noted the lack of a burnup limit requirement in the fuel criteria. There ~~is~~ a parallel fuel burnup problem area (DFSER Confirmatory Item 4.2-1) for the ABWR discussed below.

For the ABWR, GE has not directly referenced GESTAR II or the fuel criteria amendment. Instead GE has provided SSAR Appendix 4B which provides a similar set of criteria for the ABWR. Appendix 4B criteria are essentially identical to the criteria approved by the staff for GESTAR II. Since they are similar to the GESTAR II criteria, they are generally acceptable. However, some additions or restrictions are necessary. The staff review of the Appendix 4B criteria and the staff audit of the generic criteria has indicated the following restrictions are necessary for the long-term use of the criteria:

- NRC-approved analytical models and analysis procedures of General Criterion (1) to be used without further review must be limited to those referenced in GESTAR II, Revision 10, or previous revisions. Methods developed and approved in later GESTAR II revisions will not automatically apply to the ABWR and will have to be specifically reviewed and approved for ABWR use.

- Fuel burnup limits must be specified and justified based on material properties versus exposure data for each fuel type used in the ABWR and may be extended only with NRC review and approval. This was identified as DFSER Confirmatory Item 4.2-1.

In response to the staff request for burnup limit and review statements in the fuel criteria, GE ~~has~~ indicated that such a criterion is unnecessary and not a safety issue. However, they ~~have~~ proposed the following statement for the SSAR.

> Burnup limits will be specified for each fuel type used in the ABWR. The current maximum exposure limit for any GE fuel design is 70 ~~gaseous waste management system~~/metric ton of uranium (GWd/MTU) peak pellet exposure (~60 GWd/MTU rod average exposure). Any extension to this maximum exposure limit in excess of 10 GWd/MTU will be submitted to the NRC for review and approval based on the available supporting materials properties vs. exposure information and planned surveillance program. In no event will the GE fuel design maximum exposure limit required by the NRC be lower than the maximum of all exposure limits approved by the NRC for LWR fuel vendors.

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found

The staff review of this submittal ~~finds~~ that the proposed limit is higher than previously approved for GE (60 GWd/MTU, peak pellet), that an unreviewed extension of 10 GWd/MTU is excessive, and that limits approved for other vendors do not necessarily apply to GE fuel without specific review for GE. The staff considers the burnup limit a safety question and have several fuel operating concerns at burnup levels above those currently approved for BWRs (about 60 GWd/MTU peak-pellet burnup). These concerns impact normal operation, off-normal transients, and accidents.

A brief summary of the concerns are:

- No prototypical LWR operating data above about 62 GWd/MTU.
- No fuel transient data above about 46 GWd/MTU.
- Significant drop in cladding ductility observed at about 60 GWd/MTU.
- Decrease in fuel thermal conductivity and changes in other physical properties.
- Changes in LOCA rod behavior at higher burnup levels.
- Fission gas release.

Other issues that need to be addressed on a design specific basis for an extension in fuel burnup are:

- Assembly and cladding corrosion.
- Fuel rod and assembly axial growth.
- Grid spacer spring relaxation.

objections

on this subject

Since GE has provided a fuel burnup limit, the staff considers DFSER Confirmatory Item 4.2-1 resolved. However, GE has been requested to augment its proposed fuel design criteria for the ABWR to include fuel burnup limits and to indicate that these limits may be extended only with NRC review and approval. a ~~Discussions with GE have not resolved the issue~~. In the ~~latest~~ penultimate submittal by GE, ~~they have~~ omitted previous ~~objectives~~ that the staff has not imposed explicit burnup limits in the past. (They had appeared indirectly in the MAPLHGR TS maximum burnup listing, but this would effectively disappear in COLRs.) However, they proposed peak burnup limits of 70 GWd/MTU (peak pellet) which is greater than limits previously approved by the staff (60), and they a ~~have~~ also proposed that extensions to approved values which would not need staff review and approval should be at least 10 GWd/MTU, which the staff considers to be highly excessive. This ~~remains an~~ was open item F4.2-1 for the ABWR review.

Core Operating Limits Reports

Insert C

With approval of the ABWR fuel criteria, new ABWR fuel designs (or changes) satisfying the criteria would not require explicit staff review, other than that required by its use by a COL applicant for the first cycle core loading.

Similar to the presentation of the ABWR fuel design, GE has provided a specific design for the control rod. This design was used in the safety analyses of SSAR Chapters 6 and 15. GE also proposed control rod design criteria, similar in concept to those for the fuel designs, to be used as a basis for the proposed control rods or future new design submittals. Just as for the fuel design, the specified control rod design used in the ABWR safety analyses will constitute, based on the staff review and approval, an approved design that may be used by the COL applicant for first cycle without further staff review. If the COL applicant changes the design, the staff will require new submittals for review and approval.

The ABWR control rod design has, for the most part, the same geometrical and material design characteristics of those approved and used for current

Open Item F4.2-1 *Generic fuel licensing (fuel burnup limit)*

Insert C, SER page 4-4

In the final response to this open item, GE provided changes to the fuel licensing acceptance criteria of Section 4B.3(2) (j), "Submittal Supporting Accelerated ABWR Schedule-Response to Open Item F4.2-1," dated February 4, 1994, which now states that (1) fuel burnup limits will be specified for fuel used in the ABWR design, (2) the current limit for the ABWR fuel is 60 GWd/MTU rod average exposure, and (3) any extension of this limit will be submitted to the NRC for review and approval.

These changes provide an acceptable resolution of the need for burnup restrictions indicated in the staff review. The 60 GWd/MTU limit is acceptable based on the staff review of high performance data for GE fuel during the NRC audit of the fuel design process for the GE 11 fuel referenced above. The data supporting the high burnup performance that were examined during the audit included GE 8X8 fuel of the type used in the ABWR reference core. Therefore, the above item becomes Confirmatory Item F4.2-1.

configuration, (5) a suppression pool cleanup system will be employed, and (6) the combined operating license applicant will develop a program for maintaining suppression pool cleanliness.

The staff believes that the actions specified by GE are appropriate; however, they do not address the potential lack of conservatism within RG 1.82, Revision 1 due to the deleterious effect of finely fragmented insulation. Reducing the total amount of insulation within the containment would not resolve this problem; as the sizing criteria is based on correlations within the Regulatory Guide. Therefore, less insulation would lead to smaller strainers. The staff believes an acceptable resolution to this issue is to size the strainers in accordance with RG 1.82, Revision 1 but provide a factor of 3 sizing margin to account for uncertainty in the synergetic effects of strainer clogging from insulation, corrosion products, and other debris.

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6.2.2 Containment Heat Removal System

The containment heat removal system, which is an integral part of the RHR system, will consist of three redundant loops. Each loop is designed so that a failure in one loop cannot cause a failure in another. In addition, each of the loops and associated equipment is located in a separate protected area of the reactor building to minimize the potential for single failure, including the loss of onsite or offsite power causing the loss of function of the entire system. The system equipment, piping, and support structures are designed to seismic Category I criteria.

The containment heat removal system encompasses the following RHR operating modes:

- Low-Pressure Flooder (LPFL) Mode

  Following a LOCA, containment cooling starts as soon as the LPFL injection flow begins. During this mode, water from the suppression pool is pumped through the RHR heat exchangers and injected into the reactor vessel. The LPFL mode is automatically initiated by a low water level in the reactor vessel or high pressure in the drywell. In addition, each loop in the RHR system can also be placed in operation by means of a manual initiation push-button switch.

- Suppression Pool Cooling Mode

  Following a LOCA, the suppression pool cooling subsystem provides a means to remove heat released into the suppression pool. During this mode of operation, water is pumped from the suppression pool through the RHR heat exchangers and back to the suppression pool. This mode is automatically initiated, as needed, by closing the LPFL injection valves and opening the suppression pool return valves. In response to an RAI, GE indicated that the heat removal function will be initiated within 10 minutes following a LOCA. The staff found this to be sufficiently conservative and adequate to achieve the necessary containment cooling function.

Open Item F6.2.1.9-1 *Suppression pool strainer*

Insert D, SER page 6-29

This was open item F6.2.1.9-1. In response to this item, GE provided an SSAR markup of Section Appendix 6C in a letter dated February 14, 1994. To account for the uncertainty in the head loss calculations, GE will size the RHR suction strainers at least 3 times the basic strainer surface area obtained from RG 1.82, Revision 1, for all breaks except the main steam line and RCIC steam line. For breaks in the main steam line and RCIC steam line, the strainer sizing will follow the criteria in RG 1.82, Revision 1.

GE indicated that in the event of RHR strainer clogging following a break in the main steam line or RCIC steam line, the shutdown cooling mode of the RHR system is capable of removing decay heat. RHR shutdown cooling takes suction directly from, and returns to, the reactor vessel. For coolant makeup to the reactor vessel during shutdown cooling operation, the high pressure core flooder would provide the necessary coolant from the condensate storage tank.

The RCIC and HPCF suction strainers in the suppression pool are sized in accordance with RG 1.82, Revision 1, and do not include the factor of 3 in sizing criteria. The RCIC and HPCF systems preferentially take suction from the condensate storage tank. In the event of transfer of the RCIC or HPCF suction to the suppression pool and the strainers become clogged, the reactor could be depressurized using the automatic depressurization system and the RHR would then remove decay heat either through reactor injection or shutdown cooling mode.

The staff finds the strainer sizing criteria proposed by GE to be acceptable, given the existence of the shutdown cooling operation to remove decay heat in the event that the RHR suction strainers become clogged following a break in the main steamline or RCIC steamline. For other breaks, the staff concludes that sizing the RHR suction 3 times the area obtained from RG 1.82, Revision 1, will sufficiently reduce the potential for clogging. Therefore, the above open item becomes Confirmatory Item F6.2.1.9-1.

- Associated Class 1E circuits will remain with or be physically separated in the same manner as those Class 1E circuits with which they are associated;

  or

  Associated Class 1E circuits will remain with or be physically separated in the same manner as those Class 1E circuits with which they are associated, from the Class 1E equipment to and including an isolation device.

- Associated Class 1E circuits (including their isolation devices or their connected loads without isolation devices) will be subject to all requirements placed on Class 1E circuits.

- Non-Class 1E circuits powered from a Class 1E power supply will be limited to power circuits related to the FMCRDs and lighting systems.

The staff concludes that Class 1E systems, equipment, and components will be adequately protected from associated Class 1E circuits and/or non-Class 1E circuits powered through an isolation device from a Class 1E power supply. In addition, the staff concluded that sufficient independence will be maintained between redundant Class 1E systems, equipment, and components. The design therefore meets the independence and protection requirements of GDC 4 and 17 of 10 CFR Part 50, Appendix A, and is acceptable.

Verification that GE has provided the above design information in a future SSAR amendment was DFSER (SECY-92-349) Confirmatory Item 8.3.3.6-1. GE has included the above design information in Sections 8.3.3.1 and 8.3.1.1.1 of SSAR Amendment 32 which is acceptable. Therefore, this item is resolved.

Certified Design Material

DFSER (SECY-92-349) Open Item 8.3.3.6-1 was related to the design description and the ITAAC for associated Class 1E circuits. The adequacy and acceptability of the ABWR design descriptions and ITAAC are evaluated in Chapter 14.1 of this report. On the basis of this evaluation, this item is resolved.

The staff has determined that a commitment was required in the design description of electrical systems that states that non-Class 1E circuits connected to the Class 1E system shall be limited to circuits in the FMCRD and lighting subsystems. This is Open Item F8.3.3.6-1. Insert E
was

8.3.3.7 Diesel Generator Protective Relaying Bypass

Section 8.3.1.1.6.4 of SSAR Amendment 10 indicated that the following identified protective relaying will trip the diesel generator and will be retained under LOCA conditions. This relaying included the generator differential, bus differential, engine over speed, low diesel cooling water pressure (two out of two sensors), and low differential pressure of secondary cooling water (two out of two sensors). Other diesel generator protective trips will be bypassed during LOCA conditions.

Open Item F8.3.3.6-1 *Adding non-Class 1E loads to Class 1E systems*

<u>Insert E, SER page 8-59</u>

GE has included this commitment in a markup of the design descriptions of Sections 2.12.1 and 2.12.2 of the Certified Design Material dated February 4, 1994. The staff finds this acceptable. This is now Confirmatory Item F8.3.3.6-1.

Amendment 20 of the SSAR. The adequacy and acceptability of the SSAR is evaluated in Chapter 1 of this report. On the basis of this evaluation this item is resolved.

The potential for creating extremely high dose rates in the upper drywell during spent fuel handling operations and the potential for high dose rates around unshielded portions of the TIP conduit are discussed in Section 12.3.2 of this report.

12.1.3 Operational Consideration

Operational considerations regarding the implementation of a radiation protection program are outside the scope of this design certification review. The COL applicant referencing the ABWR design will address these operational considerations to the level of detail provided in RG 1.70 (Rev. 3). This was identified as DFSER COL Action Item 12.1.3-1. Amendment 23 revised Section 12.1.3 of the SSAR to identify these operational considerations as an area to be addressed by the COL applicant. GE has also included this action item in the SSAR. The adequacy and acceptability of the SSAR is evaluated in Chapter 1 of this report.

12.1.4 COL License Information

Section 12.1.4 of the DSER (SECY-91-355) identifies three issues concerning compliance with RGs 8.10 (Rev. 1) and 1.8 (Rev. 2), and procedures for keeping occupational exposures ALARA, as outside the scope of this review. This was identified as DFSER COL Action Item 12.1.4-1. Amendment 20 to the SSAR revised Section 12.1.4 to properly characterize these issues. GE has also included this action item in the SSAR. The adequacy and acceptability of the SSAR is evaluated in Chapter 1 of this report.

INSERT →
5-2

12.2 Radiation Sources

The staff has audited the contained source terms and airborne radioactive material source terms provided in Section 12.2 and Chapter 11 of the ABWR SSAR for completeness against the guidelines in RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," (Rev. 3), and against the criteria set forth in Section 12.2 of NUREG-0800 (SRP). The contained source terms are used as the basis for designing radiation protection features (including radiation shielding) and for personnel dose assessment. Airborne radioactive source terms are used in the design of ventilation systems and personnel dose assessment. The staff's review consisted of ensuring that GE had either committed to following the criteria of RGs and staff positions contained in SRP Section 12.2 or provided acceptable alternatives. In addition, the staff selectively compared source terms for specific systems against those used for plants of similar design. The staff's review indicates that source term descriptions in the SSAR are not adequate to meet the criteria of RG 1.70 (Rev. 3) and NUREG-0800.

At the current stage in the ABWR design, GE does not have the specifications for the "as-built" systems nor the "as-procured" hardware that would be available for a completed plant. Therefore, GE cannot describe the radioactive system components, which will be significant in-plant radiation sources, to the level of detail specified in RG 1.70 and the SRP. Although these

Insert B-2, SER page 12-4

In Open Item F1.9-1, the staff identified the need for GE to include a COL action item related to the use of appropriate materials in the ABWR design which would reduce the potential for personnel exposures. GE provided a submittal dated February 7, 1994, which included a markup of SSAR Section 12.3.7.4, that added a COL Action Item stating that the applicant will reduce maintenance exposure through material selection following the design commitments included in SSAR Section 12.3.1.1.2. The staff found this commitment for a COL action item to be acceptable.

The WAS staff ~~is~~ evaluating ACRS comments regarding the need for verification of fires and flooding analyses in the ITAAC for buildings. This ~~is~~ was Open Item F14.3.2-1. Insert F

Conclusions

On the basis of the above, the staff concludes that the design of the plant SSCs important to safety in the ABWR can be adequately verified by ITAAC that ensure that top-level design commitments are satisfied. Therefore, the staff concludes that the ITAAC are necessary and sufficient to provide reasonable assurance that if the inspections, tests, and analyses are performed and the acceptance criteria met, the plant systems of the design have been constructed and will operate in accordance with the design certification.

14.3.2.3.2 Reactor Systems Task Group Review

The Reactor Systems Task Group had primary review responsibility for the core reactor systems and core cooling systems in CDM Section 2.0. The group had secondary review responsibilities for those systems that could affect the operation of the core reactor systems.

The task group primarily utilized the SRP in its review of the CDM to determine the safety significance of SSCs. Other sources included applicable rules and regulations, GDCs, RGs, USIs and GSIs, NRC generic correspondence, PRA, insights from ABWR safety and severe accident analyses, and operating experience. The task group also used the draft review guidance for the design control document as an aid in its review of the systems. For selected portions of the review, the staff also utilized the regulatory guidance from the Commission related to SECY-90-016, "Evolutionary Light Water Reactor Certification Issues and Their Relationship to Current Regulatory Requirements," as modified by the Commission guidance related to SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs." In addition, the task group reviewed the Tier 1 submittals (including the design description, figures, and ITAAC) of the design using the guidelines provided in the draft review guidance for the CDM as an aid for establishing consistency and completeness.

The task group reviewed the CDM systems in a similar manner as the Plant Systems Task Group because the systems were primarily fluid systems. Thus, the group examining the systems for comprehensive and consistent treatment of the issues listed in Section 14.3.2.3.1 of this report, based on their safety significance in the respective systems. The task group found that many of the systems in this area of review were classified as safety related, and thus many of the characteristics and features of these systems were judged to have safety significance. This is reflected in a higher level of detail in the CDM for these systems.

The task group reviewed the CDM to verify that plant safety analyses, such as for overpressure protection, core cooling, transients, and anticipated transient without scram (ATWS), were adequately addressed. Consequently, the task group interacted with specialists in PRA and severe accident analyses to integrate insights into design features in the CDM. For the severe accident analyses in particular, the basis for the staff's review was the Commission guidance related to SECYs 90-016 and 93-087. For severe accident analyses, Open Item F14.3.2.3.1-1 *ACRS comments on fires and flooding design*

Insert F, SER page 14-46

GE provided proposed markups to the CDM and the supporting material in the SSAR in a letter dated February 11, 1994, that incorporated the ACRS comments. The markups provided requirements in the CDM for reconciliation analyses to be conducted for the fire and flooding design to ensure that the as-built facility is consistent with the assumptions and analyses for these issues in the design certification. The staff finds the markups acceptable. This is now Confirmatory Item F14.3.2.3.1-1.

Wetwell pressure was varied from one atmosphere to normal operating and a wetwell temperature of 95 °F was assumed along with 100 percent relative humidity. The maximum drywell to wetwell pressure differential did not exceed -0.5 psid which is less than the -2.0 psid design value.

The staff finds the revised DSIL curve acceptable because the drywell and wetwell sprays actuate simultaneously in the ABWR which eliminates the possibility of a significant pressure differential between the wetwell and the drywell at the onset of drywell sprays. This effect was confirmed by calculations performed by GE using the SHEX computer code.

Insert (H1)

18.8.4 Heat Capacity Temperature Limit

~~GE's review of certain SBO sequences showed that suppression pool temperature has the potential to exceed the EPG HCTL. During a SBO, the only injection system available to the RPV is the turbine driven RCIC system. Once the HCTL is exceeded the operator is directed to depressurize the RPV. When RPV depressurization occurs RCIC, a high pressure injection system, would become unavailable for injection and may lead to heat up of the core.~~

~~GE then submitted for staff review a revised HCTL with a low-pressure endpoint temperature of 137.7 °C instead of 103.9 °C. This upward shifting of the HCTL curve postpones RPV depressurization and would increase the availability of the RCIC. Unfortunately, this upward shift also allows temperatures exceeding saturation to exist within the suppression pool.~~

~~There are disadvantages associated with operating the suppression pool at or near saturation. An extended plume of high-quality steam was observed during sub-scale experiments performed by Chun and Sonin when the pool reached saturation temperature. The staff is concerned above the existence of large steam bubbles within the suppression pool. These large steam bubbles may drift into a relatively cooler area within the suppression pool and suddenly collapse thereby jeopardizing primary containment integrity. With the lost of the RHR pumps during a SBO there is a concern of a stratified pool is a possibility.~~

~~Another consequent to these extended plumes of steam is the reduction of the scrubbing capability of the suppression pool. This would result in a direct path from the quencher to the wetwell airspace thus effectively bypassing the suppression pool.~~

~~The staff acknowledges the value of increasing the availability of the only high-pressure injection system, RCIC, during a SBO. The staff does not believe that this increased availability is significant enough to justify operating the suppression pool at or above its saturation temperature considering the disadvantage mentioned above. The staff also believes that the fire water addition system should be available for low-pressure injection once RPV depressurization takes place.~~

~~In order for the staff to find the HCTL curve acceptable, as proposed in Amendment 32 of the SSAR, GE must demonstrate that large continuous steam plumes do not occur within the suppression pool such that the containment liner integrity could be jeopardized by the sudden unstable collapse of large steam bubbles. Large steam bubbles appeared to have been observed in a~~

Open Item F18.8.4-1 *Containment related EPGs*

Insert H1, SER page 18-39

Open Issue F18.8.4-1 consisted of two issues - the heat capacity temperature limit and the low pressure venting.

For the ABWR, GE proposes the use of a Heat Capacity Temperature Limit curve (HCTL) which would require reactor vessel depressurization beginning when the suppression pool reaches 103.9C. Increasing the allowable suppression pool temperature before reactor vessel depressurization would begin permits the operation of the RCIC system for vessel injection when all other dedicated plant systems would be postulated to fail. This could occur during a Station Blackout (SBO). This proposal raises several phenomenology issues related to hydrodynamic loads.

Concerns were raised regarding operation of the suppression pool with steam discharges from safety relief valves (SRV) or the RCIC turbine exhaust during pump operation, since suppression pool operation has traditionally been restricted to HCTL curves beginning vessel depressurization at (151F). With a steam discharge from a SRV quencher or RCIC turbine exhaust sparger at suppression pool temperatures approaching 103.9C, should a unstable steam condensation process occur, the containment liner may be subjected to an excessive buckling load from a low pressure region occurring at the containment liner/suppression pool water interface. Also, a suppression pool bypass issue arises if a steam plume extends from the quencher to the suppression pool surface. This was the HCTL part of Open Item F18.8.4-1.

To resolve the above issue on unstable collapse for an extended plume where the steam jet extends beyond the quencher condensation zone, GE relied on testing perform by Drs. Chun & Sonin and described in GE's submittal of January 20, 1994. In current generation reactors, steam discharge from a SRV quencher is condensed within a cylindrical region about the quencher arms called a condensation zone. The radius is in part a function of the amount of sub-cooling which exists within the suppression pool during a discharge, with the basis for sub-cooling being set forth in NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments."

With the proposal of permitting steam discharge in the suppression pool at higher pool temperature than what has been traditional discussed with the NRC, the staff pursued the potential consequences, as follows;

1. Potential generation of a high quality steam plume extending beyond the quencher condensation zone with the plume being ingested by the ECCS inlet piping,

2. Potential for sudden collapse and an unacceptably high condensation oscillation (CO) load, should the steam plume discussed above become sufficiently buoyant to detach from the quencher source,

3. Potential extension of a steam plume from the quencher to the pool surface, thereby leading to a pool bypass path for particle scrubbing,

4. Potential for a steam discharge from the RCIC turbine exhaust sparger, causing a CO or Chugging load higher than the CO or Chugging load for LOCA or discharge of all SRVs.

The ABWR ECCS inlet piping is located approximately 1 meter (3.3 ft) below the SRV quencher devices. The staff concluded that steam plume injection by the ECCS is not possible due to the buoyant nature of the steam plume.

In the January 20, 1994 submittal, GE presented a discussion by Dr. A. Sonin addressing the potential large steam plumes drifting into cooler region of the pool, thereby creating the initial conditions for sudden collapse of the plume. The conclusion that was reached in the above stated paper was that the cooler regions (with respect to the local temperature about the quencher which is discharging) of the suppression pool are at a low elevation and azimuthally away from the quencher. During the quencher discharge a circulatory drift motion occurs as the surrounding water is entrained into the plume. As the pool temperature increases during an extended discharge, the area about the expanding steam plume is expected to be relatively well mixed horizontally. Thermal stratification will be primarily vertical, with the highest temperature being in the warm buoyant layer near the surface and the colder temperature near the bottom of the pool. The staff finds that GE's position that a condition where the steam plume could move from a warm region of the pool to a significantly colder region to be implausible is justified based on the above stated paper and experiments performed by Drs. Chun & Sonin.

A question was also discussed concerning a steam plume extending from the SRV quencher continuously to the pool surface and creating a potential pool bypass path the wetwell airspace negating any scrubbing action by the water. This issued appears to be unfounded based on the discussion in the January 20, 1994 submittal. The argument against the notion of long continuous high quality steam plume extending to the pool surface appears unlikely due turbulence about the buoyant high velocity jet formed at the quencher hole. The turbulence caused at the plume in close proximity to the quencher entrains water into the plume from its sided causing rapid loss of plume temperature and steam volume fraction with increasing distance from the quencher. In addition, independent calculation by the staff show the wetwell airspace pressurization during pool heatup to produce sufficient pressure to maintain a minimum of 40 degrees K of subcooling in the pool based on bulk pool temperature. The staff concludes that a pool bypass is not a concern based on the proposed HTCL curve .

RCIC turbine exhaust discharge during suppression pool heat was reviewed for the potential of producing pool boundary loads which may exceed LOCA loads. This issued was raised because the turbine exhaust is discharged into the pool via a sparger which may not have the same performance features for condensing steam as a X-Quencher. GE evaluated the sparger and has determined that the potential for producing CO and Chugging loads greater than LOCA seem unlikely based on a steam mass flux of approximately 48 Kg/m2-sec. At a mass flux of this magnitude, it is unlikely that CO and Chugging loads could be produced which would be of the same magnitude as LOCA loads. In addition, the ABWR SSAR specifies a bounding asymmetric load case which assumes half the drywell vents are 180 degrees out of phase with remaining vents for chugging. Based on the asymmetric loading requirement for chugging, the low mass flux at the sparger and that the sparger design has been in use on current generation BWR's without a report failure or problem, the staff finds that the HCTL curve as drawn would not produce higher loads on the containment than LOCA loads currently assumed. Therefore, the HCTL portion of the open item becomes Confirmatory Item F18.8.4-1.

saturated pool during sub-scale experiments performed by Chun and Sonin as discussed in Dr. Sonin's paper published in Nuclear Engineering and Design (1981). The staff will find acceptable a suppression pool operated near saturation if the applicant can demonstrate that the Cross-Quencher proposed for ABWR can produce a stable steam bubble when a steam discharge could occur into a suppression pool operating near the saturation temperature. Stable steam bubble size would then be defined as that size steam bubble or group of bubbles that may drift into a cooler region of the pool and condense such that suppression pool wall pressures do not exceed those wall pressures previously defined for unstable condensation oscillation loads from SRV actuations or a LOCA.

The Low Pressure Venting issue - The following ~~items are a result of~~ the November 4, 1993, conference call with GE ~~and need to be addressed to close out this item:~~ required GE to address the following items:

1. Revise EPGs (PC/P) to show that venting is restricted to the 3 cm (2 in.) line in the drywell.

2. Address suppression pool bypass mechanism through interconnection in the ACS and show the effect on the existing bypass analysis. Ensure that no other bypass pathways exist that have not been accounted for.

3. Address containment isolation configuration of interconnection in the ACS between the wetwell and drywell. GE should justify automatic control of the ACS over a normally closed penetration ensuring containment integrity.

4. Address suppression pool level issue in EPGs relating to the wetwell to drywell interconnection level. The EPGs appear to be inconsistent with the design.

5. Address suppression pool level and pressure control EPGs for injection from sources outside of containment. The EPGs appear to direct conflicting actions in that SP/L-3.3 directs operators to stop injection from sources outside containment when the suppression pool level reaches 27.2 m (89.5 ft). Whereas, PC/P-6 directs operators to spray the containment when the water level reaches 27.2 m (89.5 ft) with use of sources external to the containment.

Insert H2

18.8.5 Primary Containment Flooding

An override statement has been placed in front of ABWR EPG Step C6-2. This step directs the operator to terminate all injection into the primary containment when drywell water level reaches the bottom of the RPV if containment radiation is greater than the CDRL and RPV water level is below the top of the active fuel.

GE stated that containment flooding is to be terminated when the drywell water level reaches the bottom of the RPV during severe accident conditions when the core has melted through the vessel and dropped to the lower drywell. Flooding is terminated to avoid covering the wetwell vent path which has the containment rupture diaphragms. The wetwell vent is located at an elevation above the bottom of the RPV.

Open Item F18.8.4-1 *Containment related EPGs*

Insert H2, SER page 18-40

Resolution of Items 1 and 5 was provided in Amendment 33. GE provided revised design information for Items 2, 3, and 4 in SSAR markups of sections Appendix 6E, 6.2.4.3.2.2.2.3, and Appendix 18B dated January 13, 1994 and February 7, 1994. For Item 2, the suppression pool bypass mechanism was shown to be insignificant when compared to the suppression pool bypass capability discussed in Section 6.2.1.1.5 of the SSAR. For Item 3, GE has provided a description of the isolation provisions. For Item 4, GE has corrected the EPGs to specify the correct water level. The staff has found the information to be acceptable. Therefore, the low pressure venting portion of the open item becomes Confirmatory Item F18.8.4-2.

19.2.3.2.2 Ex-Vessel Melt Progression

Ex-vessel severe accident progression is affected by the mode and timing of the reactor vessel failure, the primary system pressure at reactor vessel failure, the composition, amount, and character of the molten core debris expelled, the type of concrete used in containment construction, and the availability of water to the lower drywell. The initial response of the containment from ex-vessel severe accident progression is largely a function of the pressure of the RCS at reactor vessel failure and the existence of water within the reactor cavity. If not prevented through design features, risk consequences are usually dominated by early containment failure mechanisms that could result from energetic severe accident phenomena such as HPME with DCH and ex-vessel steam explosions. The long term response of the containment from ex-vessel severe accident progression is largely a function of the containment pressure and temperature due to core-concrete interaction and the availability of containment heat removal mechanisms.

At high RCS pressures, the molten core debris could be ejected from the reactor vessel in jet form causing fragmentation into small particles. The potential exists for the core debris ejected from the vessel to be swept out of the lower drywell and into the upper drywell. Finely fragmented and dispersed core debris could heat the containment atmosphere and lead to large pressure spikes. In addition, chemical reactions of the core debris particulate with oxygen and steam could add to the pressurization loads. This severe accident phenomenon is known as HPME with DCH. To prevent this phenomena, the ABWR has incorporated a reliable depressurization system to provide assurance, that in the event of a core melt scenario, that failure of the RPV would occur at a low pressure. In the event that the RPV was to fail at a high pressure, the design of the ABWR containment provides an indirect pathway from the lower drywell to the upper drywell in an effort to decrease the amount of core debris that could contribute to DCH.

Insert from P.19-56

Insert (I)

RPV failure at high or low pressure coincident with water present within the lower drywell lead to FCI with the potential for rapid steam generation or steam explosions. Rapid steam generation involves the pressurization of containment compartments from non-explosive steam generation beyond the capability of the compartment to relieve the pressure such that local overpressurization failure of the compartment occurs. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water resulting in rapid vaporization and acceleration of surrounding water creating substantial pressure and impact loads. The ABWR is designed such that there is a very low likelihood of water within the lower drywell at the time of reactor vessel failure.

The eventual contact of molten core debris with concrete in the lower drywell will lead to core-concrete interaction (CCI). CCI involves the decomposition of concrete from core debris and can challenge the containment in various mechanisms, including: (1) pressurization due to the production of steam and noncondensible gases to the point of containment rupture, (2) the transport of high temperature gases and aerosols into the upper drywell leading to high temperature failure of the containment seals and penetrations, (3) liner melt-through, (4) reactor pedestal melt-through leading to relocation of the reactor vessel and tearing of containment penetrations, and (5) the production of combustible gases such as hydrogen and carbon monoxide. CCI is affected by

Open Item F19.2.3.2.2-1 *ACRS concern with equipment tunnel protection*

Insert I, SER page 19-55

GE addressed this issue in a letter dated February 7, 1994, which proposed a new SSAR section 19.E.2.3.6. GE indicated that the equipment tunnels will be partially covered with 1.2 meters of suppression pool water at the low water level allowed by technical specifications. In the event core debris melts through the equipment tunnel, the debris will enter the suppression pool and any additional gases from the lower drywell will pass through the indicated suppression pool level. Also, a HPME results in core debris and elevated temperatures within the upper drywell. In order for the containment penetrations to withstand the elevated temperatures, the operator must actuate the containment spray system. The water from the containment spray system eventually accumulates in the suppression pool, raising the water level to provide additional water coverage of the equipment tunnels. Therefore, the suppression pool water level covering the lower section of the equipment tunnels is sufficient to preclude this potential suppression pool bypass pathway. The staff finds this acceptable. This is now Confirmatory Item F19.2.3.2.2-1.

GE had not addressed this issue in the SSAR. The staff indicated that it

many factors including the availability of water to the lower drywell, the containment geometry, the composition and amount of core melt, the core melt superheat, and the type of concrete involved.

The ABWR has incorporated several design features to mitigate the effects of CCI. These include a LDF system, an ACIWA system, basaltic concrete for the lower drywell floor, and the COPS. The LDF system provides suppression pool water to assist in cooling core debris once it has entered the lower drywell. The ACIWA system provides for both reactor vessel injection and drywell spray capability to cool core debris or control containment pressurization. Basaltic concrete protects the containment liner from melt-through and decreases the amount of non-condensible gases generated during CCI when compared to limestone-based concretes. The COPS is designed to passively relieve containment pressure to prevent gross containment failure during severe accidents when the containment pressure approaches ASME Service Level C limits. This relief pathway takes advantage of the scrubbing capability of the suppression pool to limit any offsite releases.

~~19.2.3.2.1 Equipment Tunnel Protection~~

Equipment Tunnel Bypass Mode: ~~Guidance provided within SECY-90-016 and SECY-93-087 indicated that containment structures should be protected from direct contact with core debris.~~ The equipment tunnels are located on the periphery of the lower drywell at a mid-level elevation. Core debris exiting the reactor vessel has the potential to reach the tunnels. An accumulation of core debris within the tunnels could lead to melt-through and development of a suppression pool bypass mechanism. ~~The staff~~ believed that an acceptable resolution to this issue would be for GE to provide reasonable assurance that an appreciable amount of core debris would not enter the tunnels. This ~~can~~ could be done by showing that the existing equipment within the lower drywell provides a tortuous pathway to the lower drywell periphery or providing an additional shield structure over the tunnels. or entrained from the lower drywell during a HPME

Move to 19.2-55

This was Open Item F19.2.3.2.2-1

19.2.3.3 Severe Accident Mitigative Features

19.2.3.3.1 Hydrogen Generation and Control

Generation and combustion of large quantities of hydrogen is a severe accident phenomenon that can threaten containment integrity. The major source of hydrogen generated is from the oxidation of zirconium metal with steam when the zirconium reaches temperatures well above normal operating levels. This reaction is commonly referred to as the metal-water reaction.

Research indicates that in-vessel hydrogen generation associated with core-damage can vary over a wide range. The specific amount of oxidation is dependent on a variety of parameters related to sequence progression. These include the RCS pressure, the timing and flow rate of reflooding if it occurs, and the temperature profile of the reactor core during the course of the accident sequence. In addition, ex-vessel hydrogen generation must be considered. Hydrogen is produced as a result of ex-vessel core debris reacting with steam or concrete.

Beyond design basis events can generally be categorized into in-vessel and ex-vessel severe accidents. The environmental conditions resulting from these events are more limiting than those from design bases events. The NRC established a criterion to provide a reasonable level of confidence that the necessary equipment will function in the severe accident environment for the time span for which it is needed. This criterion is commonly referred to as "equipment survivability" and is fundamentally different than equipment qualification.

19.2.3.3.7.1 In-Vessel Severe Accidents

The applicable criterion for equipment, both mechanical and electrical, required for recovery from in-vessel severe accidents is provided in 10 CFR 50.34(f).

Section 50.34(f)(2)(ix)(C) indicates that:

> Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal water reaction including the environmental conditions created by activation of the hydrogen control system.

Section 50.34(f)(3)(v) indicates that systems necessary to ensure containment integrity shall be demonstrated to perform their function under conditions associated with an accident that releases hydrogen generated from 100 percent fuel clad metal-water reaction.

Section 50.34(f)(2)(xvii) requires instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluents at all potential accident release points.

Section 50.34(f)(2)(xix) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

as of SSAR Amendment 32

These regulations collectively indicate the need to perform a systematic evaluation of all equipment, both electrical and mechanical, and instrumentation to ensure its survivability for intervention into an in-vessel severe accident. This systematic evaluation has not been performed by GE.

stated in the Advance SER that it

The staff believe~~s~~d that an acceptable resolution of this issue would entail the following:

1. GE should perform an evaluation using best-estimate means of a degraded in-vessel core damage accident that results in the reaction of a 100 percent metal-water reaction. The basis for the evaluation should be included. The evaluation should identify the most likely sequences resulting in substantial oxidation of the fuel cladding as a result of the probabilistic safety assessment. An example of an acceptable sequence would involve accident conditions where ECCS performance was degraded for a sufficient period of time to cause cladding oxidation but

is later recovered to ensure a safe shutdown. If the analysis assumes an intact primary loop, the basis for this should be supported by the results of the PSA (i.e., LOCA does not contribute significantly to core melt). The impact on the reactor system and containment system from the pressure, temperature, and radiation released should be evaluated. As an example, the safe shutdown and containment equipment identified below should be evaluated. Plots showing pressure and temperature as a function of time should be provided.

In the event that the in-vessel severe accident environment has no effect on the equipment performance, this should be clearly indicated along with the supporting rationale. Examples of such instances include cases where the equipment has already performed its function prior to the onset of the accident conditions or the equipment is located in an area not exposed to the environmental conditions, such as being located outside of primary containment. For equipment where environmental conditions as a result of the in-vessel severe accident are in excess of the equipment qualification range, an engineering rationale must be developed as to why the equipment would survive the environment for the needed time span. This rationale could include such factors as: limited time period in the environment; the use of similar equipment in commercial industry exposed to the same environment; the use of analytical extrapolations; or the results of tests performed in the nuclear industry or at national laboratories.

An acceptable example using this rationale is the work that GE performed for electrical penetration assemblies in Section 19F.3.2.2 of the SSAR. In particular, GE referenced experimental tests performed at Sandia National Laboratories on actual electrical penetration assemblies (EPAs) used in operating plants. The tests were performed at representative severe accident conditions with temperatures up to 371 °C (700 °F) and pressures up to 965 kPa (140 psig). Using the results of this work, GE committed to providing EPAs which will maintain leak tightness up to containment pressure of 924 kPa (134 psig) and a temperature of 371 °C (700 °F). The end result of this is that the assumptions used for equipment performance in GE's severe accident evaluation are consistent with the as-built plant.

Safe shutdown equipment that should be addressed include: Scram Equipment, HPCF motor & pump, HPCF isolation valves, HPCF controls, RCIC turbine & pump, RCIC Steam Valves & cables, RCIC controls, RHR, ADS, Shutdown Cooling, etc.

Equipment for containment integrity should include: Containment Structure, CIVs - inboard, CIVs - outboard, Electrical Penetrations, Mechanical Penetrations, Hatches, Sealing Mechanisms (welds, bellows, O-ring), etc.

2. With respect to instrumentation requirements, the staff believes that sufficient instrumentation should exist to inform operators of the status of the reactor and the containment at all times as the in-vessel severe accident is intended to be recoverable from and lead to safe shutdown

with containment integrity maintained. The emergency operating procedures (EOPs) direct specific manual operator actions based on instrumentation readings and as such all instrumentation should exist where manual operator actions are specified within the EOPs. As a minimum, the instrumentation identified below should be evaluated.

The instrumentation is designed to survive the environment as specified in RG 1.97. However, RG 1.97 only ensures that the instrumentation will survive in the worst environment resulting from a design bases event and not a severe accident. Therefore, engineering rationale must be developed as to why the instrumentation would survive the environment. This rationale could include such factors as: limited time period in the environment; the use of similar equipment in commercial industry exposed to the same environment; the use of analytical extrapolations; or the results of tests performed in the nuclear industry or at national laboratories.

Instrumentation should include: Neutron Flux, RPV Water Level, RPV Pressure, Sup Pool Temp., Sup Pool Level, DW/WW H2 Conc, DW/WW O2 Conc, DW Temperature, DW Pressure, WW Pressure, WW Temperature, DW Water Level, etc.

Insert J1 →

19.2.3.3.7.2 Ex-Vessel Severe Accidents

The applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents is provided in the Equipment Survivability section of SECY-90-016. This section indicates that features provided only (not required for design basis accidents) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements, 10 CFR Part 50, Appendix B quality assurance requirements, and 10 CFR Part 50, Appendix A redundancy/diversity requirements. The reason for this judgement is that the staff does not believe that severe core damage accidents should be design basis accidents in the traditional sense that DBAs have been treated in the past.

However, 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 cases where safety related equipment (equipment provided for DBAs) is relied upon to cope with severe accident situations, there should be reasonable assurance that this equipment will survive accident conditions for the period that is needed to perform its intended function.

According to SECY-90-016, GE was to review the various severe accident scenarios analyzed and identify the equipment needed to perform its function during a severe accident and the environmental conditions under which the equipment must function. Equipment survivability expectations under severe accident conditions should include consideration of the circumstances of applicable initiating events (e.g., SBO, earthquakes) and the environment (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function. The staff conclude~~s~~ that GE ~~has~~ not performed the evaluation as outlined by SECY-90-016.
d had
as of Amendment 32 to the SSAR

Open Item F19.2.3.3.7-1 *Equipment Survivability*

<u>Insert J1, SER page 19-80</u>

In response to the open item, GE provided the environmental profiles, a table of the necessary equipment, and the accompanying rationale for in-vessel severe accidents in an SSAR markup of Section 19E.2.1.2.3 dated February 7, 1994. The staff finds this information acceptable. The above open item becomes Confirmatory Item F19.2.3.3.7.1-1.

As stated in The Advance SER

The staff believes that an acceptable resolution of this issue would entail the following:

1. GE should provide an evaluation of the dominant accident sequences identified in Section 19E.2.2 of the SSAR. For each accident sequence, GE should identify the mitigation features. Mitigation features should include ADS, ACIWA, and RCIC as appropriate.

   In addition, the specific environment profile (pressure, temperature, radiation fields) should be specified. For each mitigation feature an assessment of survivability should be done using ground rules similar to those specified above for in-vessel accident. At least the following mitigation features should be evaluated: SRVs, Containment Structure, Vacuum Breakers, CIVs - Inboard, CIVs - Outboard, Electrical Penetrations, Mechanical Penetrations, Hatches, Sealing Mechanisms (welds, bellows, O-rings), Passive Flooders, COPS, COPS CIVs, etc.

2. With respect to instrumentation requirements, the staff believes that sufficient instrumentation should exist to inform operators of the status of the containment at all times, and of the reactor during the early stages of the accident to ensure reactor failure at low pressure or allow for low pressure injection from the AC independent water addition system.

   As a minimum, the list of instrumentation identified below should be evaluated. Where extended ranges of operation of the instrumentation is needed, it should be identified along with the environment to which the instrumentation will be exposed.

   The instrumentation is designed to survive the environment as specified in RG 1.97. However, RG 1.97 only ensures that the instrumentation will survive in the worst environment resulting from a design bases event and not a severe accident. Therefore, engineering rationale must be developed as to why the instrumentation would survive the environment. This rationale could include such factors as: limited time period in the environment; the use of similar equipment in commercial industry exposed to the same environment; the use of analytical extrapolations; or the results of tests performed in the nuclear industry or at national laboratories.

   At least the following instrumentation should be evaluated: RPV Water Level, RPV Pressure, Sup Pool Temp., Sup Pool Level, DW/WW H2 Conc, DW/WW O2 Conc, DW Temperature, WW Pressure, WW Temperature, etc.

Insert J2 J3

19.2.3.3.8 Protection of Containment Sumps

The lower drywell contains two sumps - an equipment drain sump (EDS) and a floor drain sump (FDS). Figures 1.2-3b and 1.2-13e of the ABWR SSAR indicate that the sumps are embedded in the lower drywell floor with dimensions of approximately 1 m (3 ft) wide, 2 m (7 ft) long, and 1.25 m (4 ft) deep. Figure 1.2-3b of the ABWR SSAR indicates that the lower drywell has approximately 1.6 meters of concrete protecting the liner, while SSAR Section 6.2.1.1.10.3 indicates that there is 1.5 meters (5 ft) of concrete protecting the liner. Therefore, in the sump region, there is approximately 0.25 to 0.35 m (1 ft) of concrete protecting the containment liner. An

Open Item F19.2.3.3.7-1 *Equipment Survivability*

<u>Insert J2, SER page 19-81</u>

In response to the open item, GE provided the environmental profiles, a table of the necessary equipment, and the accompanying rationale for ex-vessel severe accidents in an SSAR markup of section 19E.2.1.2.3 dated February 7, 1994. The staff finds this information acceptable. The above open item becomes Confirmatory Item F19.2.3.3.7.2-1.

Open Item F19.2.3.3.7-1 *Equipment Survivability*

Insert J3, SER page 19-81

19.2.3.3.7.3 Basis for Acceptability

GE developed a set of curves representing the bounding environmental conditions for both in-vessel and ex-vessel severe accidents. The environmental conditions were then compared to the equipment capabilities to provide a measure of confidence that the necessary equipment would survive the expected conditions. The staff concludes that the systematic process used by GE for assessing equipment survivability is acceptable and consistent with the assumptions used in GE's deterministic severe accident assessment.

accumulation of core debris within the sumps could lead to accelerated core-concrete interactions and given the decreased thickness of concrete protecting the containment liner, the time to liner melt-through in the sump region from core-concrete interactions could be adversely affected.

To prevent liner melt-through in the sump region, the ABWR will have a protective layer of refractory bricks (corium shield) built around each sump to prevent corium ingression. The corium shield design is discussed in Sections 6.2.1.1.10.4 and 19ED of the ABWR SSAR.

19.2.3.3.8.1 Sump Design Criteria

The following general criteria included in Amendment 32, were developed by GE for designing the sumps:

- Corium shield height greater than maximum height of core debris bed.

- Melting point of corium shield material above initial contact temperature.

- Corium shield material to have good chemical resistance to siliceous slags and reducing environments.

- Seismic adequacy determined during detailed design phase.

The EDS and FDS have different functions and therefore specific design criteria in addition to the above GDC were developed. The specific design criteria is discussed below.

19.2.3.3.8.1.1 Equipment Drain Sump

The purpose of the EDS is to collect water leaking from valves and piping within the containment. The water enters and exits through piping from above the sump. As such, the following additional design criteria ~~have been~~ were specified ~~.~~ in Amendment 32:

- Solid corium shield, except for the inlet and outlet piping through the roof.

- Corium shield walls thick enough to withstand ablation.

- Corium shield placed directly on top of lower drywell floor.

19.2.3.3.8.1.2 Floor Drain Sump

The purpose of the FDS is to collect water which falls onto the lower drywell floor. The water flows across the drywell floor and runs into the FDS at a height equal to the lower drywell elevation. As such, the following additional design criteria ~~have been~~ were specified ~~.~~ in Amendment 32:

- Corium shield will have channels at the lower drywell elevation to allow for water collection during normal operation.

- Channel ~~height~~ length ensures that debris will freeze before reaching the sump.

- Width and number of channels ensure required water flow rate during normal operation.

- Corium shield walls thick enough to allow residence time for debris solidification within the channels.

- Corium shield will extend beneath the lower drywell floor.

- Corium shield height to ensure long term debris solidification.

19.2.3.3.8.2 Corium Shield Design

IN Amendment 32

In Section 6.2.1.1.10.4.2, GE indicated that the EDS corium shield is made of alumina with a height of 0.4 m (1 ft). ~~GE did not provide the thickness of the EDS corium shield necessary to withstand ablation.~~

in Amendment 32

For the FDS corium shield, GE analyzed the ability of the corium shield to initially freeze molten debris as it enters the channels and to transfer sufficient heat such that the debris remains solid in the long term. This analyses ~~is~~ was in Sections 19ED.4 and 19ED.5 of the ABWR SSAR. This analysis was used along with the GDC and the specific design criteria specified above to determine the actual corium shield design. The FDS corium shield, as specified in Section 6.2.1.1.10.4.2 of the ABWR SSAR, is made of alumina, with a height of 0.4 m (1 ft). The channels in the FDS corium shield are 1 cm (.4 in.) high and 1.06 m (3.48 ft) long, however, the width has not been specified. The shield extends 0.4 m (1 ft) below the channels.

in Amendment 32

in Amendment 32

19.2.3.3.8.3 Discussion

Stated in the Advance SER that it

in Amendment 32 was

The staff believes that the sump shield designs proposed by ~~GE~~ have considerable merit and that some conservatism exists in the specified design criteria. For example, the design criteria is intended to ensure that no core debris enters the sumps. However, in actuality, the sumps could withstand limited amounts of core debris. In addition, GE did not take credit for flooding the lower drywell with the LDF system or AC independent water addition system. Based on engineering judgement, the staff believes that the sump shields would prevent a substantial accumulation of core debris and that the channels within the FDS would lead to freezing of debris within them. However, the analysis provided to support the proposed shield designs ~~is~~ not sufficient to reach this conclusion. In particular, GE did not make use of existing experimental data and analytical tools in justifying their design. This was Open Item F19.2.3.3.8.2-1.

In the Advance SER stated

The staff ~~believes~~ that an acceptable resolution to this issue would entail the following:

1. GE should evaluate related experimental and analytical work performed in this area to lend additional credibility to their design. In particular, GE should address how the results of the previous work supports their design. This would include a discussion on the prototypically of the core debris, important parameters and results. The staff has performed a quick review of related work in this area and believes that it is relevant and readily available.

a. Experiments performed at (1) KfK on ingression of molten debris into small cracks and openings, (2) Winfrith in the United Kingdom, and (3) Grenoble in France.

b. Analytical tools such as PLUGM (NUREG/CR-3190) and BUCOGEL (CEA).

c. Work performed in forging and casting industries.

2. The analysis performed by GE in Amendment 32 ~~for sizing the FDS~~ evaluated an oxidic melt of around 2500k (4040 °F) and a eutectic melt of around 1700k (2600 °F). However, GE used the same correlations and key parameters for both, such as thermal conductivity and latent heat of fusion. To account for uncertainty in the progression of a severe accident and a range of material properties (density, melting point, thermal conductivity, etc.), GE should perform separate analysis for an oxidic, metallic, and eutectic melt clearly identifying the material properties and providing suitable references. In addition, GE should identify the parameters that the shields are most sensitive to (i.e., freezing point, heat of fusion, velocity of debris in channel, atmosphere temperature, melt superheat, etc.). GE can use the results of their MAAP runs to identify the core debris composition at the time it enters the lower drywell. In addition, GE could use the results of other code predictions (BWRSAR, MELCOR) as documented in NUREGs for similar BWRs.

3. GE should address the following; why the velocity of the debris in the FDS channels does not have to consider the initial velocity of debris from falling from the reactor pressure vessel (RVP)?

4. GE should modify the design criteria to:

   a. Specify that the EDS extends below the lower drywell floor and that both shields prevent tunneling of core debris under them.

   b. Specify sloping of the shield roof to prevent accumulation of core debris or show that the long-term debris solidification in the channels is not affected by minor amounts of debris of the roof.

5. GE should provide the thickness of the EDS corium shield necessary to withstand ablation.

Insert K

19.2.4 Containment Performance

The NRC approach for ensuring containment survivability from severe accident challenges consists of requiring inclusion of accident prevention and consequence mitigation features and the containment performance goal (CPG). The CPG ensures that the containment would perform its function in the face of most severe accident challenges and that the design (including its mitigation features) would be adequate if called upon to mitigate a severe accident.

Two alternative CPG were identified in SECY-90-016 - 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 for containment performance during a severe-accident challenge would be appropriate for the evolutionary LWRs in place of a CCFP.

Open Item F19.2.3.3.8-1 *Containment sump design*

<u>Insert K, SER page 19-84</u>

GE has addressed these 5 issues in an SSAR markup of Sections 6.2.1.1.10.4 and Appendix 19ED dated February 7, 1994. specifically, GE provided a markup of SSAR Section 19ED.6 providing an overview of related experimental and analytical work concerning the freezing of molten fuel in narrow channels to address item 1. For item 2, GE modified the analysis of channel length in Subsection 19ED.4 to account for three debris scenarios covering the expected range of melt phenomena. For item 3, GE clarified the FDS shield design to clarify the location of channels. For item 4, GE modified subsections 19ED.3 and 19ED.5.2 and added 19ED.5.3 to establish that the EDS and FDS shields extend to the sump floor to prevent debris tunneling. Further, GE modified its long term analysis in Subsection 19ED.5.1 to credit flooding of the lower drywell. Lastly, for item 5, GE added subsection 19ED.5.3 to address the thickness of the EDS shield walls and the shield wall of the FDS without channels. In addition GE modified its general and specific design criteria for the EDS and FDS. This resulted in a revision to the design dimensions. Further, GE takes credit for the lower drywell flooder in determining the sump shield design. The staff concludes that the sump shield design proposed by GE is acceptable. This is based on GE's development of design criteria, proposed analytical solution, evaluation of the shield design to variations in key parameters, and review of existing related experimental and analytical work. Therefore, the above open item becomes Confirmatory Item F19.2.3.3.8.3-1.

The staff considers this a COL action item and will ensure that COL applicants provide guidance on controlling and maintaining the integrity of freeze seals. The guidance should address the use of an engineering safety analysis on a case-by-case basis to ensure that the use of freeze seals, where a failure could result in loss of inventory, will not result in any unresolved safety review questions. This is COL Action Item 19.3.3.2.1

Reactor Internal Pump Motor and Impeller Replacement

The ABWR RIPs are used to supply coolant circulation and to replace the external coolant recirculation system used in the BWR designs. This is a design improvement over the BWR designs in which an unisolable pipe break or component repair in the external recirculation system could result in a major loss of inventory control. The RIP concept was adopted from European BWRs which have been operated for over 15 years and have had no indications of difficulty in maintenance or in operation that resulted in a loss of inventory. The applicant discussed the procedures to maintain and to replace the RIPs in SSAR Section 19.Q.4.2 of the ABWR PRA Shutdown Risk Final Report.

Removal of the RIP motors for maintenance is accomplished by using integral inflatable seals which act as backup sealing devices to assure that no RPV water leakage occur. Following each motor removal a temporary cover plate is bolted to the bottom of the motor's housing which forms part of the reactor vessel. The impeller is then removed from the top. Upon the removal of the impeller, the bolted cover plate acts as the RPV boundary and prevents leakage of reactor water. A plug is then installed on the RPV bottom head at the impeller nozzle to provide additional protection against draining the RPV.

The staff asked the applicant to discuss the effect the increased temperature from a loss of DHR cooling during RIP replacement or maintenance could have on inflatable seals. In a letter of January 13, 1993, the applicant stated that the inflatable seals on the RIP shaft will be permanently installed and will be designed to handle normal operating temperatures. Increased temperatures from a loss of DHR cooling will not affect the performance of the seals since the coolant temperature during shutdown conditions will be less than the design temperature of the seal.

The staff reviewed the RIP maintenance and replacement sequences, and found that if leakage occurred during the removal of the RIP motor, the temporary cover plate could be installed to eliminate RPV water leakage and the pump internal primary and inflatable secondary seals would minimize the potential RPV water drainage, and the RIP motor removal process, therefore, acceptable. However, the staff ~~find~~s noted that during the RIP impeller and shaft removal, possible unisolated LOCA with an opening of about 20.32 cm (8 in.) exists in the event that operators failure to follow the maintenance procedure or possibly as a result of miscommunication. In addition, during pump impeller and shaft replacement, the containment would be opened, thus allowing a direct release path to the environment. The staff, therefore, requires that RIP impeller and shaft replacement be conducted only after fuel has been removed from vessel. ~~The staff considers this an open item.~~ This ~~is~~ was Open Item F19.3.3.2.1-1.

Insert (L)

Open Item F19.3.3.2.1-1 *RIP maintenance and replacement*
Insert L, SER page 19-101

In a letter dated January 14, 1994, and subsequent responses to staff questions, GE provided additional information with regard to RIP impeller-shaft removal and CRD replacement. During the RIP shaft and impeller removal, the following replacement sequence, maintenance requirements, and pump design features together with the refueling platform auxiliary hoist design are intended to minimize the likelihood of an unisolable LOCA.

- Upon completion of the RIP motor removal, a maintenance cover is bolted to the bottom of the motor housing, which forms a temporary RPV boundary. The motor housing is then pressurized to verify that the maintenance cover is providing a seal. The secondary inflatable seal is then depressurized. At this point, two seals (internal primary metal-to-metal, and maintenance cover) are still provided. Upon removal of the pump impeller-shaft, only the maintenance cover seal remains. To protect against removed of the impeller-shaft in the event that maintenance seal is not in place, an auxiliary hoist interlock is provided. The refueling platform auxiliary hoist interlocks will interrupt the hoisting power if the load exceeds a specified setpoint. The hoist load setpoint is less than the sum of the impeller-shaft weight and the hydrostatic head on the impeller. To over come the static head, the motor housing must be pressurized, which requires the maintenance cover plate to be secured in place, and thus sealing is assuring.

- When the pump impeller-shaft has been removed, a maintenance diffuser plug is then installed over the shaft opening. The diffuser plug provides sealing and is the only means to prevent possible unisolable LOCA when the motor housing is drained and the maintenance cover plate is removed for secondary inflatable seal and stretch tube inspection or replacement. To prevent this potential unisolable LOCA, the diffuser plug is designed with a break-away lifting lug. If the maintenance cover is not secured in place and pressurized, the lifting lug will break during the attempted removal due to the static head pressure exceeding the lug's design force, thus ensuring that the diffuser plug seal is maintained. In the event that the operator inadvertently removed the plug, abnormal or excessive drainage will be discovered when the motor housing is partially drained through the drain line. At this point, RIP sealing is still provided by the maintenance cover plate. Discontinue drainage of the motor housing will eliminate the loss of reactor coolant and allow corrective actions.

In SSAR Section 5.4.15.4, GE provided a markup to state that the COL applicant shall develop procedures to ensure appropriate installation and verification of motor bottom cover, as well as visual monitoring of the potential leakage during impeller-shaft and maintenance plug removal. In addition, the COL applicant shall develop a contingency plan (e.g. close personnel access hatch, safety injection), which assures that core and spent fuel cooling can be provided in the event that a loss of coolant occurs during RIP maintenance. This is acceptable and this item is now Confirmatory Item F19.3.3.2.2-1.

was

Control Rod Drive Replacement

CRD replacement for the ABWR is similar to current BWRs, and will use the same maintenance procedures. The CRD is withdrawn to the point where the CRD blade back seats onto the CRD guide tube. This provides a metal-to-metal seal that minimizes the RPV water drainage when CRD is removed. The staff reviewed the replacement process and found that unisolated LOCA with an opening of about 5.08 cm (2 in.) exist at the bottom of the vessel head if the CRD blade and drive simultaneously removed due to operator failures to follow the procedures. It ~~is~~ the staff position that TSs should be included to prohibit the removal of the blade and drive of the same assembly. ~~The staff considers this an open item~~. This ~~is~~ was Open Item F19.3.3.2.1-2.

Insert M

19.3.3.2.2 Alternate Reactor Inventory Control Feature

The ABWR design includes the non-safety-related feedwater and condensate system, consisting of three electric driven pumps and associated piping, that can be used as an alternate means for make-up during shutdown operation. The CRD pump also can be used to provide inventory control during shutdown by injecting water from the condensate storage tank to the RPV through the FMCRD system. An ACIWA system is also available to supply make-up water to the RPV if no ECCS make-up water is available.

The staff finds these provisions acceptable and concludes that the applicant has sufficiently addressed the concerns in NUREG-1449 related to alternate make-up capability to provide core decay heat removal. The alternate inventory control features using the feedwater, the condensate system and the CRD pump will provide alternate core cooling upon loss of normal RHR capability. The staff also finds that an ACIWA system will further enhance the capability of the ABWR to maintain core cooling in the event that no ECCS makeup is available.

19.3.3.3 Containment Integrity

During refueling of the ABWR, the primary containment head is removed and cannot be readily repositioned to restore containment integrity. This is also the case for operating BWR plants with Mark I and II containments. NUREG-1449 indicated that BWR secondary containments were judged unlikely to prevent an early release following initiation of boiling with an open RCS or during potential severe-core-damage scenarios. This is also the case for the ABWR.

In NUREG-1449, the staff evaluated the need to re-establish containment integrity for all operating plants under shutdown conditions. Based on operating experience, thermal-hydraulic analyses, and PRA assessments, it was concluded that containment integrity under some shutdown conditions may be necessary for PWR plants. However, this conclusion was not reached for BWR plants. This is due in part to the decreased frequency and significance of precursor events involving reduction in reactor vessel level or loss of RHR (or both) in BWRs as compared to PWRs. In addition, BWRs do not enter a midloop operating condition as do PWRs.

In NUREG-1449, staff stated that operating BWR alternate DHR methods provide significant depth and diversity. For these reasons, the staff concluded that loss of RHR in BWRs during shutdown is not a significant safety issue as long

Open Item F19.3.3.2.1-2 *CRD maintenance and replacement*
Insert M, SER page 19-102

GE stated that the CRD blade normally remains in this backseated condition at all times with the FMCRD out. In the event that the CRD blade is required to be removed for replacement, a temporary blind flange will be first installed on the end of the CRD housing to prevent draining of the reactor water.

During the FMCRD removal, personnel are required to monitor under the RPV for water leakage out of the CRD housing. If abnormal or excessive leakage occurs after only a partial lowering of the FMCRD, which is the indicative of a metal-to-metal seal that has not yet been established, the FMCRD can then be raised back into its installed position to eliminate the leak and allow corrective action. In the event that the CRD blade and drive of the same assembly were inadvertently removed due to operator failures to follow procedures during refueling operations (water level greater than 23 ft above the vessel flange), the analysis results indicated that it would take approximately 36 minutes for the water to fill the lower drywell sump, reach the tunnel entrances and begin flowing into the access tunnels. With the expected flow rate of 174 cubic meters per hour (6,145 ft^3/hr) from the CRD opening, the water in the spent fuel would drop approximately .3 m/hr (.98 ft/hr). The high drywell sump level and the low spent fuel level would alarm in the main control room approximately 2 minutes and 28 minutes, respectively, into the transient. The normally operating non-safety-related makeup water condensate system (MUWC) will automatically start upon receiving a low level alarm in the spent fuel pool and transfer water to the spent fuel pool cooling and cleanup (FPCCU) system. The RHR spent fuel cooling mode can be manually initiated to provide makeup injection and the suppression pool clean up system also can provide backup if the MUWC is not available. In the event of loss of off-site power, backup water also can be provided by RHR AC independent water addition system.

Upon identified leakages from the bottom of the RPV, it is expected that personnel door and equipment hatch in the lower drywell areas will be closed within 30 minutes before the water level would reach the tunnel entrances and begin flowing into the access tunnel. Appropriate actions will then be taken to reinsert the CRD blade and to mitigate the event using various water sources and injection systems as mentioned.

The staff also notes that only two or three complete FMCRDs are required to be removed for inspection each refueling outage. This is an improvement relative to the CRD system design at current BWRs which have piston seal replacement needs such that 20 to 30 drives are typically removed each refueling outage.

In SSAR Section 4.6.6.1, GE provided a markup to state that the COL applicant shall develop procedures to ensure that maintenance procedures have provisions prohibit coincident removal of the CRD blade and drive of the same assembly. In addition, the COL applicant shall develop contingency procedures to provide core and spent fuel cooling capability and mitigative actions during CRD replacement with fuel in the vessel. The staff determined that GE's proposed SSAR changes are acceptable and that no TS changes are necessary. Therefore, the above open issue is now Confirmatory Item F19.3.3.2.3-1.

SSAR Section 1A.2.23 states that the ABWR high leak detection and isolation system processes the differential pressure signals that isolate the RCIC turbine. Spurious trips are avoided because the RCIC has a bypass startup system controlled by valves F037 and F045. Upon receiving RCIC start signals, bypass valve F045 opens to pressurize the line downstream and accelerate the turbine. The bypass line through F045 is small (diameter of 1 in) and naturally limits the initial flow surge to prevent a differential pressure spike in the upstream pipe.

After approximately 5 to 10 seconds, steam supply valve F037 opens to admit full steam flow to the turbine. At this stage, the line downstream is already pressurized. This design feature will reduce the possibility that a pressure spike would occur during any phase of the normal startup process. In the DFSER, the staff concluded that the ABWR design adequately addresses the requirements of this TMI item. However, the staff indicated that the COL applicant should test the RCIC bypass startup system during plant startup and designated this as DFSER [COL] Action Item 20.3.1-4. ~~GE has not included a COL action item in the SSAR addressing this test, therefore, COL Action Item 20.3.1-4 will remain open until GE has done so.~~ Insert B-3

20.4.65 Issue II.K.3(16): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Reduction of Challenges and Failures of Relief Valves; Feasibility Study and System Modification

Refer to the evaluation of 10 CFR 50.34(f)(1)(vi) in Section 20.5.6 of this report.

20.4.66 Issue II.K.3(17): Measures to Mitigate Small-Break Loss-of-Coolant Accidents and Loss-of-Feedwater Accidents -- Final Recommendations of Bulletins and Orders Task Force - Report on Outage of ECC Systems; Licensee Report and Technical Specification Changes

This TMI item required all GE plants to review data on ECC system outages to determine if cumulative outage time limitations should be incorporated in TS. It also required submittal of a report detailing outage dates, lengths of outages, and causes of the outages for all ECCSs.

The DFSER reported that the STS permit several components of the ECCS to have substantial outage times (e.g., 72 hours for one diesel generator; 14 days for the HPCI system). The TS do not specify cumulative outage time limitations for ECCSs. This was identified in the DFSER as TS Item 20.3.1. This was not required to be in the ABWR TS, but will be implemented in Plant Administrative procedures as discussed in Section 16.x of this report.

In the DFSER, the reported that SSAR Section 1A.3.5 included a requirement for the COL applicant to report ECCS outages in annual summary reports to the NRC. The staff also report that it would review compliance with this requirement during the COL review. This was identified in the DFSER as COL Action Item 20.3.1-5. The staff has verified that SSAR Section 1A.2.5 includes a COL action item to prepare and submit an annual report on ECCS unavailability that also includes the required information discussed above. This is approach is acceptable.

Open Item 1.9-1 *Additional COL Action Items*

Insert B-3, SER page 20-87

In a letter dated February 7, 1994, GE provided a proposed revision of SSAR Section 1A.2.23 and new Section 1A.3.8 that establish a COL action item for the COL applicant to test the RCIC bypass startup system during plant startup. This is acceptable to the staff and DFSER COL Action Item 20.3.1-4 is resolved contingent on incorporation of the changes in the final SSAR.

A reactor head vent line is a continuous vent which is normally open to discharge to a main steamline.

Insert B-4

The COL applicant will develop plant-specific procedures to govern the operator's use of the relief mode for venting the reactor. This was identified in the DFSER as COL Action Item 20.3-1. ~~GE has not included a COL action item in the SSAR addressing the development of these procedures, therefore, COL Action Item 20.3-1 will remain open until GE has done so.~~

GE has submitted no additional accident analyses to address a break in any of the vent lines because the plant's design basis includes a complete steamline break, which is more bounding.

The staff concurs with the applicant's assessment because it includes adequate capacity, operation, and procedural provisions of the ABWR vent system. The staff concludes that GE has adequately addressed the requirements of this TMI item.

20.5.19 10 CFR 50.34(f)(2)(vii): Consideration of Degraded or Melted Cores in Safety Review - Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation (TMI Item II.B.2)

Paragraph (2)(vii) of 10 CFR 50.34(f) requires radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment. GE's response adequately addresses the requirements of this TMI item as discussed in Sections 12.3.6 and 13.6.3.5 of this report.

20.5.20 10 CFR 50.34(f)(2)(viii): Consideration of Degraded or Melted Cores in Safety Review - Post-Accident Sampling (TMI Item II.B.3)

Paragraph (2)(viii) of 10 CFR 50.34(f) requires 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 exposure to any individual exceeding 5 rem to the whole-body or 75 rem to the extremities.

GE's response adequately addresses the requirements of this TMI item as discussed in Section 9.3.2.2 of this report.

20.5.21 10 CFR 50.34(f)(2)(ix): Consideration of Degraded or Melted Cores in Safety Review - Rulemaking Proceeding on Degraded Core Accidents (TMI Item II.B.8), "Hydrogen Control System"

Paragraph (2)(ix) of 10 CFR 50.34(f) requires a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel clad metal-water reaction. The hydrogen control system and associated systems shall provide with reasonable assurance that:

Open Item 1.9-1 *Additional COL Action Items*

Insert B-4, SER page 20-117

In Amendment 33, GE revised SSAR Sections 1A.2.5 and 1A.3.6 to establish a COL action item for the COL applicant to develop the indicated plant procedures. This is acceptable to the staff and DFSER COL Action Item 20.3-1 is resolved.

However, the staff required GE to discuss GL 92-04 in the SSAR and include any design changes necessary to preclude the potential for false reactor coolant level readings. The staff also required GE to determine if compliance with this TMI requirement was affected by any design changes. This was identified in the DFSER as Open Item 20.3-8.

The known common-mode deficiencies in BWR level instrumentation systems have been addressed at operating BWRs and by GE in the ABWR design. It should also be noted that these particular deficiencies would not have compromised the automatic protective functions of the level instrumentation for accident scenarios initiated while at power, and that no previous incidents at BWRs of inaccurate level indication have been misinterpreted by plant operators so as to lead to unsafe actions. In view of the importance of level instrumentation for safety in BWRs, and the experience discussed above where the potential existed to fail redundant level instruments due to a common cause, the staff believes that the addition of level instrumentation which operates on a diverse physical principle is desirable and prudent for the purpose of guiding operator emergency actions. The staff ~~concludes~~ concluded that the ABWR level instrumentation system without the proposed level diversity meets the minimum requirements of all applicable GDC.

in the Advance SER

GE ~~does~~ did not agree with the staff recommendation for diverse water level instrumentation and ~~has~~ presented its position in a letter dated October 26, 1993. As part of the letter, GE presented the following summary:

> ABWR water level instrumentation is rugged, simple and highly redundant for failure tolerance. All known operating problems have been addressed in this design and it is incredible to postulate simultaneous common-mode failures which would yield identical errors in all the dp instrumentation. Alternate technologies are unqualified for this application; further, there is no need to add this complexity, since the plant operating staff has ample additional indications of an impending problem without relying solely on water level. The EPGs direct the operator to use all information available to him and make conservative (safe) decisions.

In the attachment to the letter, GE also provided a list of indications of inadequate RPV water level which are independent of the dp RPV water level instrumentation. The staff recognizes that other parameters could aid the operator in assessing the adequacy of core cooling under accident conditions. These include instrumentation for indication of reactor power, core neutron flux, the recirculation flow control system response, and feedwater flow and steam flow mismatch. However, the staff believes that these indications could be easily misinterpreted or could be insufficient because they are only indirect methods of inferring reactor water level or core cooling.

Other evolutionary designs, such as the ABB-Combustion Engineering (CE) System 80+, provide diverse methods of RPV level measurement. The inadequate core cooling instrumentation package in the CE System 80+ plant includes reactor vessel level monitoring system probes employing both dp sensors and the heated junction thermocouple concept. The staff is aware of a diverse method of level monitoring that is currently in use in at least one nuclear power plant in Germany employing ultrasonic measurement techniques. In addition, a diverse level measurement system which uses heated junction thermocouples has been in use for the past five years at a Swedish BWR, and another Swedish BWR uses float switches for diverse level indication and automatic systems actuation. Other Swedish BWRs have decided in principle to install diverse level measurement systems.

The staff indicated in the Advance SER that

The diverse method of level measurement is recommended for indication in the control room only (there is diverse instrumentation, namely high drywell pressure, in both the operating BWRs and the ABWR design which provides diverse signals for automatic safety systems actuation for many event scenarios). This would provide a direct and back-up means for the operator to identify inadequate core cooling and to take appropriate manual actions to initiate and control safety systems as identified in the plant emergency operating procedures. The staff recommend~~s~~ that the diverse level measurement device be reliable, redundant, and capable of being powered by on-site power sources.

also ed

F20.5.30-1

[illegible] was Open Item ~~20.3-6 remains unresolved until the RPV level diversity issue is resolved~~.

Insert N

In the DFSER the staff reported that GE indicated that the human factors aspects of this requirement are beyond the scope of the ABWR design certification review and the COL applicant will be responsible for addressing them in the detailed design implementation. This was identified in the DFSER as COL Action Item 18.7.2.2-3. The staff has verified that GE established a COL action item (Item 18.8.1) in SSAR Section 18.8 for the detailed control room development as defined in DD Table 3.1 ITAAC and in SSAR Section 18E. Further, GE has established a COL action item (Item 18.8.4) in SSAR Section 18.8 to address II.F.1. This approach is acceptable to the staff as discussed in Section 18.7.2.2 of this report.

20.5.31 10 CFR 50.34(f)(2)(xix): Instrumentation and Controls - Instruments for Monitoring Accident Conditions (TMI Item II.F.3)

Paragraph (2)(xix) of 10 CFR 50.34(f) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

SSAR 7.5 compares the ABWR design against the criteria of RG 1.97, Revision 3, addressing accident monitoring instrumentation. Section 7.5 lists the variables that are considered essential safety-related information for the operators, and identifies specific exceptions to the guidance of RG 1.97. The list incorporates adequate monitoring capability for post-accident plant conditions that include core damage, including reactor pressure, water level and temperature, containment pressure, temperature and radiation level, and shutdown operation status. Based on its review of SSAR Section 7.5, the staff has concluded that the ABWR I&C design meets RG 1.97 as discussed further in Section 7.5.2 of this report and, therefore, also meets this TMI requirement.

Open Item 20.5.30-1 *RPV Water Level Instrumentation Diversity*

Insert N, SER page 20-125

The staff issued the draft Commission paper, "Diversity in the Method of Measuring Reactor Pressure Vessel Level in the Advanced Boiling Water Reactor and Simplified Boiling Water Reactor" on November 15, 1993, for public and industry comments. The ACRS discussed the issue in its 404th meeting on December 9-11, 1993, and sent its recommendation to the Commission in a letter dated December 16, 1993. The ACRS did not support the staff recommendation on diversity. Based on ACRS deliberations and GE's position, the staff reconsidered the need for the requirement for instrumentation diversity.

All the known common-mode deficiencies in BWR level instrumentation systems have been addressed by GE in the ABWR design. It should also be noted that these deficiencies would not have compromised the automatic functions of the level instrumentation for accident scenarios initiated while at power, and that no previous incidents at BWRs of inaccurate level indication have been misinterpreted by plant operators so as to lead to unsafe actions. In addition, for many events, the ECCS is started in ABWR on high drywell pressure, as well as low reactor water level, thus providing some diversity. The ABWR EPGs will be used to develop the EOP that will be used with the reactor water level instrumentation.

Even though it may be desirable to provide instrumentation diversity in the ABWR design, there is not sufficient basis to postulate an unidentified potential common-mode failure. Further, diverse level measurement devices have not been demonstrated to be adequate. In light of the enhanced LOCA response in the ABWR and the guidance provided in the ABWR EPGs to address the use of the RPV instrumentation, the staff concludes that diversity is not required for the ABWR. On the basis of the above discussion, Open Item F20.5.30-1 (DFSER Open Item 20.3-8) is resolved.