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Security Notice

This letter forwards Security-Related Information to be withheld in accordance with 10 CFR 2.390. Upon removal of Enclosure 3, the balance of this letter may be considered non-Security-Related.

Docket No. 52-001

MFN 10-342
December 7, 2010

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **ABWR Standard Plant Design Certification Renewal Application Design Control Document, Revision 5, Tier 1 and Tier 2**

GE Hitachi Nuclear Energy ("GEH") is pleased to request NRC renewal of the ABWR standard plant design certification (10 CFR Part 52, Appendix A) and requests approval of an accompanying amendment to the ABWR Design Control Document ("DCD"), Tier 1 and Tier 2. Revision 5 of the ABWR DCD incorporates information associated with a change in the applicant, a containment reanalysis, and an aircraft impact assessment. GE Nuclear Energy (predecessor of GEH) was the original applicant for the ABWR certified design. Accordingly, GEH requests that the NRC retain the original ABWR certified design as a separate appendix in 10 CFR Part 52 and issue a separate appendix for other applicants seeking to amend the certified design. GEH discusses in Enclosure 1 the policy and safety basis for this request.

The six enclosures in this transmittal contain information related to the renewal request and the revised DCD files:

- Enclosure 1 contains an explanation of the content of the application and identifies related documents that may be provided under separate cover.

- Enclosure 2 contains an Introduction to the DCD, Revision 5.

- Enclosure 3 contains the Tier 1 and Tier 2 documents with "Security-Related Information" protected from public disclosure under the provisions in 10 CFR 2.390. The security-related information is marked with the designation "{{{Security-Related Information – Withhold Under 10 CFR 2.390}}}." GEH requests that the NRC withhold from public disclosure this information in accordance with the provisions of 10 CFR 2.390.

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- Enclosure 4 contains public versions of the Tier 1 and Tier 2 from which the security-related information has been withheld (*i.e.*, redacted).

- Enclosure 5 contains the original GE Nuclear Energy Report 25A5680, "SAMDA Technical Support Document for the ABWR", Revision 1, as submitted to the NRC in MFN 162-94, December 21, 1994 (NRC Accession Number 9503290339; ML100210563). This GENE report is being provided to the staff for convenience as a historical reference since the report was originally submitted in 1994.

- Enclosure 6 contains the "Supplemental Environmental Report - Amendment to ABWR Standard Design Certification," GEH Report 25A5680AA, required by NRC regulation 10 CFR 51.55(b) for an amendment to a design certification.

If you have any questions, please contact me.

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

I declare under penalty of perjury that the foregoing is true and correct.

Executed on the 7th day of December 2010.

Jerald Head

Jerald G. Head
Senior Vice President, Regulatory Affairs
GE-Hitachi Nuclear Energy Americas LLC

Enclosures:

1. Description of the ABWR Design Certification Renewal Application
2. CD – ABWR DCD, Revision 5, Introduction
3. CD – ABWR DCD, Revision 5, Tier 1 and Tier 2 w/Security-Related Information
4. CD – ABWR DCD, Revision 5, Tier 1 and Tier 2 w/o Security-Related Information (Public Version)
5. SAMDA Technical Support Document for the ABWR, Revision 1
6. CD - Supplemental Environmental Report - Amendment to ABWR Standard Design Certification

cc: w/enclosures 1, 2, 3, 4 , 5 and 6

A. Muniz, NRC

cc: w/o enclosures 2, 3, 4 and 6

A. Cubbage, NRC
J. Head, GEH
H. Madronero, GEH
C. Reda, GEH
D. Roderick, GEH

Enclosure 1

MFN 10-342

Description of the ABWR Design Certification Renewal Application

Enclosure 1

Description of the ABWR Design Certification Renewal Application

In accordance with 10 CFR 52.57(a), GE Hitachi Nuclear Energy (GEH) requests the NRC renew the Advanced Boiling Water Reactor (ABWR) design certification (10 CFR Part 52, Appendix A). The ABWR design certification rule, effective June 11, 1997, would otherwise expire at the end of a period of fifteen years, or June 11, 2012. In addition, in accordance with 10 CFR 52.59(c), GEH requests NRC approval of an amendment to the DCD, as explained further below.

Background

GE Nuclear Energy, predecessor of GEH, was the original applicant for the ABWR design certification. The basic ABWR design was developed by a consortium of GE Nuclear Energy, Hitachi, and Toshiba, and has been used in plants constructed in Japan. The NRC-certified ABWR design is based on U.S. codes and standards that were in effect at the time of certification. Currently, GEH is involved in two projects in Taiwan, Lungmen Units 3 and 4, for which the design is based on the NRC-certified ABWR design. In addition, the Combined License application for STP Units 3 and 4 references the ABWR design certification, with an alternate vendor supplying the design. The STP application contains a significant number and a broad scope of modifications to the certified design with which GEH is not involved.

Applicant

The renewal applicant is GEH, which is a designer of nuclear reactors, a supplier of nuclear services, and a manufacturer of nuclear fuel. GEH is majority owned and controlled by a U.S.-based company, General Electric, in alliance with Hitachi, a Japanese company. GEH conducts business internationally, with headquarters in Wilmington, North Carolina. Technical qualifications of GEH are included in the application documents, as described below. The application has been prepared in accordance with the GEH Quality Assurance Program.

Standardization

In 1987, the Commission issued its Policy Statement on Nuclear Power Plant Standardization. The Policy Statement encourages the use of standard plant designs and provides information concerning certification of plant designs that are essentially complete in scope and level of detail (52 Fed. Reg. 34884 (Sept. 15, 1987)). The Policy Statement explains that the intent of these actions described therein "are to improve the licensing process and to reduce the complexity and uncertainty in the regulatory process for standardized plants." The Policy Statement concludes that the use of certified designs in future license applications should enhance and benefit public health and safety, as well as contribute to stability and predictability in the regulatory process. Consistent with the Policy Statement goals, the NRC developed the 10

CFR Part 52 design certification process as a rulemaking to ensure, among other things, a high level of finality for a standard design in license applications (*see* 73 Fed. Reg. 15372, 15373 – 15378 (April 18, 1989)). Regulations in 10 CFR Part 52, and in the ABWR design certification rule, limit changes to a standard design in the interest of standardization. In addition, regulations for renewal of a design certification support maintaining the standard design as originally certified, unless (1) the applicant requests an amendment, or (2) the NRC must impose changes for assuring adequate protection. Note, however, that an applicant also must address the requirements of 10 CFR 50.150 regarding aircraft impact assessment (see 10 CFR 52.59 ¶¶ (a), (b), and (c)).

The original ABWR design certification was the first action completed under 10 CFR Part 52 and, while benefits of a standard design certification are only beginning to be demonstrated, GEH remains committed to the overall concept of standardization. GEH is, therefore, maintaining essentially intact the original design information upon which the NRC based its original certification decision, with a limited scope of proposed changes. GEH discussed this approach with the NRC in a pre-application meeting February 23, 2010. In response to questions during the meeting, GEH explained a strategy that may include a future request for a more extensive amendment to the ABWR design certification that would incorporate departures in a future Combined License application. Such a strategy better supports NRC regulations and guidance for maintaining standard designs from several perspectives:

- First, an amendment requested as part of a request for renewal must meet NRC regulations at the time of renewal (10 CFR 52.59(c)). By maintaining essentially intact the original design information as part of the renewal, with only limited proposed changes, GEH minimizes potentially conflicting regulatory requirements in the certified design.

- Second, GEH is not currently involved in a U.S. reference plant project using the certified ABWR design and, thus, an extensive amendment requested as part of a renewal application at this time may not represent the set of changes that a U.S. customer may ultimately request (note that GEH does not consider the STP Units 3 and 4 Combined License application as a reference for a GEH ABWR design due to the significance and number of departures that incorporate Toshiba-Westinghouse detailed design (including certain design changes), in which GEH has not been involved and for which GEH cannot validate as consistent with the licensing basis for the original ABWR design, as certified).

- A third consideration for limiting the scope of changes to the ABWR certified design is that NRC regulations include a provision that if an amendment requested as part of a renewal application entails such an extensive change to the design certification that an essentially new standard design is being proposed, an application for a design certification must be filed in accordance with 10 CFR Part 52 (10 CFR 52.59(c)).

Further supporting standardization, GEH, as successor of the original applicant, requests that the NRC incorporate no other design changes submitted as part of another entity's amendment request into the ABWR certified design rule. Because GEH continues to support the viability of the ABWR certified design through existing international projects and potential U.S. projects, it would be inappropriate for the NRC to disadvantage the original applicant of a design certification by approving multiple, and potentially conflicting, changes that could result in an internally conflicting certified design and one that relies on proprietary information of multiple entities, thereby making it a design that no entity could fully support as a standard design.

Such a result is contrary to NRC's Policy Statement on standardization in that it could result in a decrease in safety of the certified design. Accordingly, GEH requests that the NRC maintain the original ABWR certified design as a GENE/GEH design by treating any other entity's amendment request either as an essentially new standard design, depending upon the extent of proposed changes (see 10 CFR 52.59(c)), or as a similar but separate ABWR design certification in a new appendix to 10 CFR Part 52. Such action also is consistent with the spirit of the Commission's original intent that a vendor whose design is certified through rulemaking would have protection (through NRC regulations, the Administrative Procedure Act, and judicial review) "against arbitrary amendment or recission of the certification rule" (73 Fed. Reg. at 15375).

Regulatory Compliance

NRC regulations for renewal of a design certification specify that a renewal may be granted if the design, either as originally certified or as modified during the rulemaking on the renewal, complies with the Atomic Energy Act and the Commission's regulations applicable and in effect at the time the certification was issued, provided, however, that the first time the Commission issues a rule granting the renewal for a standard design certification in effect on July 13, 2009, the Commission shall, in addition, find that the renewed design complies with the applicable requirements of 10 CFR 50.150 (see 10 CFR 52.59(a)). As noted above, GEH addresses the requirements of 10 CFR 50.150 regarding aircraft impact assessment in the application.

Table 1 below provides an assessment of the application as compared to NRC regulatory requirements for renewal of a design certification application. NRC regulations do not require that the design certification meet regulations in effect at the time of renewal, except as related to associated amendments. Therefore, the current requirements in 10 CFR 52.47, "Contents of Applications, Technical Information," are not addressed in Table 1.

GEH reviewed the items listed in 10 CFR 52.47 for any that might relate to the proposed amendment to the DCD. GEH addresses below under "Application Documents" (1) technical qualifications of GEH as the applicant (10 CFR 52.47(a)(7)), and (2) conformance to the Standard Review Plan section applicable to the containment analysis (10 CFR 52.47(a)(9)). GEH acknowledges that additional information may be addressed or reconciled in future Combined License applications that reference the ABWR certified design or in a future amendment request for the ABWR certified design rule.

In addition, 10 CFR 52.47 ¶¶ (a)(21) and (a)(22) require discussion of resolution of certain generic issues and incorporation of operating experience into the design. Although these requirements are not specifically applicable to the proposed amendment included in the renewal application, information on these two requirements is discussed further below.

Application Documents

Applicant: GEH is the applicant of the renewal request. Changes to the original design certification are minimized, as described herein and in the spirit of standardization, as discussed above. Aside from the limited scope changes, GEH has, however, identified those specific references to GE Nuclear Energy and changes each instance, as appropriate, to refer to GEH as the applicant. GEH considers this overall change as administrative and discusses it no further for purposes of this application, except to the extent that the technical qualifications of GEH are described in the application documents.

Design Control Document: NRC regulations in 10 CFR Part 52, Subpart B, set forth the type of information that must be included in a design certification application. Because GEH is submitting an application for renewal of the previously certified ABWR design, the application contains a complete final safety analysis report in the form of a Design Control Document (DCD) that describes the standard design information. Regarding the specific amendments included in this renewal application, GEH submits Revision 5 of the ABWR DCD in total to reflect the changes, as marked with revision bars on the right side of the affected pages (note that the design certification was based on Revision 4 of the DCD). The specific sections that include proposed changes are identified in a change list that accompanies the DCD.

The DCD changes are limited to addressing technical requirements of GEH as the new entity applicant (see Section 1.4), correcting the containment peak pressure analysis to reflect a more limiting line break that GEH identified and discussed in MFN 09-306 (June 8, 2009, ML100640164), and incorporating the results of an assessment of aircraft impact according to requirements in 10 CFR 52.59(a) and 10 CFR 50.150. A public version of the DCD and a SUNSI version of the DCD, which contains certain security-related information that is withheld from disclosure from the public, are provided in electronic media.

DCD Introduction: GEH also includes a copy of the DCD Introduction that was submitted with the original certified design material. The DCD Introduction includes information that explains the process used in the design certification and in development of the DCD. For example, the DCD Introduction includes a list of the sections of the DCD that contain information determined by the NRC to be Tier 2* information during the original design certification rulemaking activity.

Applicant's Supplemental Environmental Report – Amendment to Standard Design Certification: According to NRC regulations in 10 CFR51.55(b)[1], an amendment to a design

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[1] **§ 51.55 Environmental report—standard design certification.**

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(b) Each applicant for an amendment to a design certification shall submit with its application a separate document entitled, "Applicant's Supplemental Environmental Report - Amendment to Standard Design Certification." The environmental report must address whether the design change which is the subject of the proposed amendment either renders a severe accident mitigation design alternative previously rejected in an environmental assessment to become cost beneficial, or results in the identification of new
(footnote continued)

certification must include a supplemental Environmental Report (ER). Because GEH is proposing amendments to the design certification, a supplemental ER applies to the application. Although GEH determined that the original Severe Accident Mitigation Design Alternative (SAMDA) assessment is not impacted by the proposed changes to the DCD, the supplemental ER discusses the design changes and provides the necessary explanations and justifications for this conclusion. The original SAMDA Technical Support Document for the ABWR was submitted to the NRC in MFN 162-94, December 21, 1994 (NRC Accession Number 9503290339; ML100210563). A copy of the original SAMDA report (Enclosure 5) is provided for reference along with the supplemental ER (Enclosure 6).

Conformance with Standard Review Plan: For the proposed changes to the DCD associated with the containment peak pressure reanalysis, the results of the reanalysis and the bases for the changes are discussed in Revision 5 of the DCD. NUREG-0800, "Standard Review Plan" (SRP), Section 6.2.1.1.C, "Pressure-Suppression Type BWR Containments" (Rev. 7, March 2007), addresses analytical models that are acceptable for calculating the containment peak pressure and temperature. The SRP states for the ABWR:

> *For ABWR plants, the calculated results for containment short-term and long-term response to postulated line breaks are based on the General Electric Mark III (ABWR) analytical model that was used in the ABWR standard plant analysis evaluated by the NRC in the ABWR FSER.*

GEH has followed the analytical modeling and has described the assumptions in Revision 5 of the DCD. The SRP also discusses margins in terms of a construction permit and at the operating stage. The DCD is essentially the basis for a license, so the DCD describes the results in terms of the margin to the design values for demonstrating conformance to the SRP:

> *For BWR pressure-suppression plants at the operating license stage, the peak calculated containment pressure and differential pressure should be less than the design values.*

Refer to Revision 5 of the DCD, Section 6.2 and Appendix 3B, for indications of changes associated with the containment reanalysis.

SUNSI Proprietary Information and Security-Related Information: Proprietary information associated with the original design certification review was submitted as part of the Standard Safety Analysis Report (SSAR). Because of the nature of the information, it was not incorporated into the DCD, but remained as part of the SSAR. This information has not been revised as part of amendments in Revision 5 of the DCD. Accordingly, the information is not resubmitted as part of the renewal application. GEH is, however, prepared to submit the information in a separate transmittal, if the NRC requests.

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severe accident mitigation design alternatives that may be reasonably incorporated into the design certification.

In the February 23, 2010, pre-application meeting, GEH informed the NRC that it planned to submit a Technical Report regarding the containment reanalysis and containing any associated proprietary information. GEH has determined that sufficient non-proprietary information is included in the DCD. Therefore, no Technical Report is submitted as part of the renewal application. In addition, NRC previously suspended its review of GEH ABWR Licensing Topical Reports (LTRs) that were associated with changes to the DCD as part of the STP Units 3 and 4 projects. The review should remain suspended, as these LTRs are not related to the renewal application.

No additional proprietary information is associated with the amendment included as part of the renewal application. Certain information is considered SUNSI-Security-Related Information and is redacted from the DCD consistent with the redactions in Revision 4 posted on the NRC website. GEH considered this the best approach so as to avoid any inconsistencies in the treatment of sensitive information.

Probabilistic Risk Assessment (PRA): GEH has not revised the PRA that was performed for the original certified design. Therefore, there are no changes to Chapter 19, which describes the results of the PRA. Amendments to Chapter 19 are, however, included to address the aircraft impact assessment. Specific changes are identified in the change list accompanying the DCD.

Safeguards Information: Certain changes are reflected in the safeguards information that was part of the original certified design. No changes to the safeguards information are proposed for the amendment. Accordingly, that information is not re-submitted to the NRC. However, GEH is prepared to provide this information if the NRC so requests.

Restricted Data/Classified Information: No restricted data or classified information is applicable to the ABWR certified design information.

Operating Experience and Generic Issues: Regarding operating experience, GEH discussed its review of 10 CFR Part 21 with regards to the ABWR in MFN 09-306 (June 8, 2009, ML100640164). As described in the letter, the containment analysis was in error. The proposed amendment included with the renewal application addresses the containment reanalysis that has been performed to address the previous error.

Table 2 below addresses certain generic issues according to guidance in NRC Regulatory Guide (RG) 1.206, Section C.IV.8, Guidance.[2] As shown in Table 2, GEH describes where and

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[2] RG 1.206, C.IV.8: "Appendix B to NUREG-0933 lists generic issues that are applicable to future reactor plants. The applicant should address those items in Appendix B to NUREG-0933 that are designated USI, HIGH, MEDIUM, NOTE 1 (possible resolution identified for evaluation), NOTE 2 (resolution available (documented in NUREG, NRC memorandum, SER, or equivalent)), CONTINUE, and NOTE 6 (new requirements for future plants recommended). The agency resolved those items in Appendix B marked by NOTE 3(a) or I with an effective date for future plants by establishing new regulatory requirements and/or positions (rule, regulatory guide, SRP change, or equivalent). Therefore, if the application addresses these items elsewhere, it need not address them again under the requirements of 10 CFR 52.79(a)(20).
(footnote continued)

how the generic issues were addressed as part of the original design certification in the DCD and identifies those that would need to be addressed at a later time in a Combined License application or in a future DCD amendment outside of the renewal scope. No changes to the DCD are proposed at this time to address any of these generic issues.

Applicants should address those generic issues for which the "Future Plants Effective Date" column includes either no entry or "TBD."" GEH NOTE: The information in Table 2 is focused on the items within this specific scope.

Table 1. Regulatory Compliance

NRC Regulatory Provision of 10 CFR	GEH ABWR Design Certification Renewal Application
§ 52.55 Duration of certification.	
(a) Except as provided in paragraph (b) of this section, a standard design certification issued under this subpart is valid for 15 years from the date of issuance. *(b) A standard design certification continues to be valid beyond the date of expiration in any proceeding on an application for a combined license or an operating license that references the standard design certification and is docketed either before the date of expiration of the certification, or, if a timely application for renewal of the certification has been filed, before the Commission has determined whether to renew the certification. A design certification also continues to be valid beyond the date of expiration in any hearing held under § 52.103 before operation begins under a combined license that references the design certification.* *(c) An applicant for a construction permit or a combined license may, at its own risk, reference in its application a design for which a design certification application has been docketed but not granted.*	The original ABWR design certification expires June 11, 2012, if not renewed, but may remain valid under timely renewal once an application for renewal is filed and before the Commission has made a decision on the renewal request. The GEH application should trigger timely renewal while the NRC review proceeds.
§ 52.57 Application for renewal.	
(a) Not less than 12 nor more than 36 months before the	Because the original ABWR design certification expires June 11, 2012, an application for renewal must be filed within the period of

NRC Regulatory Provision of 10 CFR	GEH ABWR Design Certification Renewal Application
expiration of the initial 15-year period, or any later renewal period, any person may apply for renewal of the certification. An application for renewal must contain all information necessary to bring up to date the information and data contained in the previous application. The Commission will require, before renewal of certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit if this information is necessary for the Commission to make its safety determination. Notice and comment procedures must be used for a rulemaking proceeding on the application for renewal. The Commission, in its discretion, may require the use of additional procedures in individual renewal proceedings.	June 11, 2009, and June 11, 2011. The GEH application falls within this time period. GEH has determined two issues to bring up to date the information and data contained in the original certified Design Control Document, Revision 4, according to (1) GEH identification of an error in the containment peak pressure analysis (see MFN 09-306 (June 8, 2009, ML100640164) and (2) NRC requirements of 10 CFR 50.150 to address aircraft impacts in a design certification renewal application as follows: *10 CFR 50.150 (a)(3)(iii)(B) Renewal of standard design certifications in effect on July 13, 2009 which have not been amended to comply with the requirements of this section by the time of application for renewal.* Regarding the level of detail of information, GEH addresses these two issues consistent with the level of detail in the original certified design information submitted on the docket. Revision 5 of the Design Control Document contains updated information regarding the two issues described above and the supporting engineering information is available for audit by the NRC during its review of the application. GEH recognizes that the NRC rulemaking process will be used to renew the ABWR design certification and that this process will include a notice and comment procedure, as well as other procedures that the Commission may require.
(b) A design certification, either original or renewed, for which a timely application for renewal has been filed remains in effect until the Commission has determined whether to renew the certification. If the certification is not renewed, it continues	As noted above, GEH submits its application for renewal within the time period prescribed by NRC regulations. Thus, the application should be subject to timely renewal.

NRC Regulatory Provision of 10 CFR	GEH ABWR Design Certification Renewal Application
to be valid in certain proceedings, in accordance with the provisions of § 52.55.	
(c) The Commission shall refer a copy of the application for renewal to the Advisory Committee on Reactor Safeguards (ACRS). The ACRS shall report on those portions of the application which concern safety and shall apply the criteria set forth in § 52.59.	GEH understands that an ACRS review will be conducted during the NRC review process for the renewal application.
§ 52.59 Criteria for renewal.	
(a) The Commission shall issue a rule granting the renewal if the design, either as originally certified or as modified during the rulemaking on the renewal, complies with the Atomic Energy Act and the Commission's regulations applicable and in effect at the time the certification was issued, provided, however, that the first time the Commission issues a rule granting the renewal for a standard design certification in effect on July 13, 2009, the Commission shall, in addition, find that the renewed design complies with the applicable requirements of 10 CFR 50.150.	The GEH application includes changes related to a reanalysis of the containment peak pressure. For this issue, there are no changes in regulatory requirements applicable to this issue. Therefore, the application complies with this provision of the regulations. Regarding the aircraft impact assessment required by 10 CFR 50.150, GEH has included applicable changes to the Design Control Document to describe results of the assessment in accordance with 10 CFR 50.150. Thus, the application complies with this provision of the regulations.
(b) The Commission may impose other requirements if it determines that: *(1) They are necessary for adequate protection to public health and safety or common defense and security;* *(2) They are necessary for compliance with the Commission's regulations and orders applicable and in effect at the time the design certification was issued; or* *(3) There is a substantial increase in overall protection of the*	GEH has addressed issues that it considers necessary to amend the Design Control Document for maintaining adequate protection to the public, to address changes to NRC regulations that impose additional requirements on the renewal of a design certification. GEH has not proposed additional changes and, thus, the remainder of the Design Control Document maintains the NRC conclusions on the original certified design.

NRC Regulatory Provision of 10 CFR	GEH ABWR Design Certification Renewal Application
public health and safety or the common defense and security to be derived from the new requirements, and the direct and indirect costs of implementing those requirements are justified in view of this increased protection.	
(c) In addition, the applicant for renewal may request an amendment to the design certification. The Commission shall grant the amendment request if it determines that the amendment will comply with the Atomic Energy Act and the Commission's regulations in effect at the time of renewal. If the amendment request entails such an extensive change to the design certification that an essentially new standard design is being proposed, an application for a design certification must be filed in accordance with this subpart.	GEH requests amendment to the Design Control Document to include the two issues described above. These two issues represent minimal changes and do not represent "such an extensive change" to the Design Control Document that a new standard design is being proposed. While GEH noted above that the containment peak pressure reanalysis complies with NRC regulations that were in place at the time of certification, as required by 10 CFR 52.59(a), the amendment also complies with current applicable NRC regulations. GEH expects that the applicable regulations will remain the same during the NRC review of the application. However, if the NRC amends those regulations during the time period of its review, GEH will review such amendments to determine if any further changes are necessary. Regarding the requirements in 10 CFR 50.150, the application addresses applicable requirements for renewal of a design certification, as described above.
(d) Denial of renewal does not bar the applicant, or another applicant, from filing a new application for certification of the design, which proposes design changes that correct the deficiencies cited in the denial of the renewal.	GEH does not expect that the NRC will deny the request for renewal. However, if any deficiencies in the proposed design changes are identified, GEH expects that the NRC will discuss any such deficiencies in the proposed design changes with GEH (through the NRC process for requesting additional information following acceptance of the application) so that GEH may correct those in order for NRC to proceed with its review.

NRC Regulatory Provision of 10 CFR	GEH ABWR Design Certification Renewal Application
§ 52.61 Duration of renewal.	
Each renewal of certification for a standard design will be for not less than 10, nor more than 15 years.	GEH requests that the NRC renew the ABWR design certification for the full 15 years.
10 CFR Part 52, Appendix A	
VII. Duration of This Appendix *This appendix may be referenced for a period of 15 years from June 11, 1997 except as provided for in 10 CFR 52.55(b) and 52.57(b). This appendix remains valid for an applicant or licensee who references this appendix until the application is withdrawn or the license expires, including any period of extended operation under a renewed license.*	This section is directly from the current ABWR design certification rule 10 CFR Part 52, Appendix A. It is consistent with the requirements discussed above and does not impact content of the DCD.

Table 2. ABWR Review -- NUREG-0933, Appendix B,

"Applicability of NUREG-0933 Issues to Operating and Future Reactor Plants"

Regulatory Guide 1.206, Section C.IV.8, Guidance

Review of Items Marked With No Entry or as "TBD"[3]

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
TMI Action Items I.A through III.D.4	TMI Action Plan Items	The TMI action items were resolved during the original design certification, as described in the DCD. See Tier 2, Section 1A, of ABWR DCD.
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	This GSI is not applicable to ABWR.
A-25	Non-Safety Loads on Class 1E Power Sources	This item was resolved in the ABWR DCD Tier 2, Section 19B.2.9, in the original design certification: The ABWR design assures the reliability and safety of the Class 1E power sources and safety-related systems by a highly selective connection (i.e., only one subsystem) of nonsafety-related equipment and strict control of the interface between this subsystem and Class 1E power system. Each safety related system conforms to the requirements of IEEE Standard384 (Reference 19B.2.9-2) and meets RG 1.75 (Reference 19B.2.9-3) and addresses IEEE Standard 279 (Reference 19B.2.9-5). The ABWR design incorporates three independent Class 1E diesel generators (DGs) and a non-Class 1E combustion turbine generator (CTG). The CTG is designed to automatically and independently assume the plant investment protection (PIP) loads, should a LOPP [loss of preferred power] event occur. This is in much the same

[3] Note that the information is largely extracted from the referenced section of the DCD.

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
		manner as the DGs assume the Class 1E loads for the same event. Therefore, it is not necessary for the Class 1E buses to assume the PIP loads. (See Subsections 8.2.1 and 8.3.1.) The ABWR design excludes non-Class 1E from the Class 1E buses, with the exception of the fine-motion control rod drive (FMCRD) subsystem, the associated AC standby lighting system, and the associated DC emergency lighting system. The reliability of the FMCRD subsystem is enhanced for the anticipated transient without scram (ATWS) event by using Class 1E power for the drive motors. Class 1E load breakers in the switchgear are part of the isolation scheme between the Class 1E power and the non-Class 1E FMCRD loads. In addition to the normal overcurrent tripping of these load breakers, zone selective interlocking (ZSI) is provided between them and the upstream Class 1E bus feed breakers. The Class 1E load breakers, in conjunction with the ZSI feature, provides the needed isolation between the Class 1E bus and the non-Class 1E loads. (See Subsection 8.3.1.1.1 for more details on this feature relative to the FMCRD power circuits.) Since both the safety systems and their Class 1E power supplies conform to the requirements of IEEE Standard 384 and meet the intent of Regulatory Guide 1.75, an acceptable level of safety exists for both the safety systems and their Class 1E power supplies. Therefore, this issue is resolved for the ABWR.
A-35	Adequacy of Offsite Power Systems	This item was resolved in the ABWR DCD Tier 2, Section 19B.2.11, in the original design certification: The conceptual design of an offsite power system and station switchyard(s) for the ABWR design is given in Section 8.2. The interface requirements will ensure that the switchyard(s) provide redundant offsite power feed capability to the nuclear unit, consisting of two preferred power circuits, each capable of supplying the necessary safety loads and other equipment. The ABWR onsite power systems are described in Section 8.3, and include three redundant and independent 6.9kV Class 1E safety buses. The incoming source breakers trip upon loss of normal power, and emergency power is provided to each Class 1E bus by separate and independent diesel

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
		generator (DG) units. A combustion turbine generator automatically assumes the plant investment protection loads, but can be used to manually provide back-up power for any Class 1E bus, should a DG fail or be out of service. The Class 1E AC Power Systems are described in Subsection 8.3.1.1. Protection against degraded voltage is specifically addressed in Subsection 8.3.1.1.7(8). The protection schemes are designed according to the recommendations of IEEE Standard 741 (Reference 19B.2.11-3), which is consistent with the guidance of BTP PSB-1. The ABWR Standard Plant Class 1E auxiliary power system is designed in compliance with General Design Criterion (GDC) 18 (Reference 19B.2.11-4) so that inspection, maintenance, calibration and testing can be carried out with a minimum of interference with operation of the nuclear unit, as described in Subsection 8.3.1.1.5.3. On-line testing is greatly enhanced by the design, which utilizes three independent Class 1E divisions. Indication of the system unavailability is provided in the control room. A Technical Specification establishes limiting conditions for operations, surveillance requirements, trip setpoints with minimum and maximum limits, and allowable values for the undervoltage protection sensors and associated time delay devices. Protection of the Class 1E power supplies to safety-related equipment from the effects of an undervoltage condition of the offsite power source thus conforms to the guidance of BTP PSB-1, and this issue is, therefore, resolved for the ABWR Standard Plant design.
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation	This item was resolved in the ABWR DCD Tier 2, Section 19B.2.21, in the original design certification: The filter systems required to perform safety-related functions following a design basis accident are the standby gas treatment system (SGTS) and the control room habitability system as described in Sections 6.4 (Habitability Systems) and 9.4.1.1 (Control Room Habitability Area HVAC), and Subsection 6.5.1 (Engineered Safety Features Filter Systems). The SGTS consists of two parallel and redundant filter trains. Each filter train is designed to have a HEPA filter installed at both inlet and

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
	Systems	outlet sides of the charcoal adsorber. The CRHA HVAC system is provided with redundant divisions. Each division consists of an emergency filtration unit. A HEPA filter is also provided before and after the charcoal adsorber of each emergency filtration unit. The HEPA filters of these systems will be tested periodically with DOP using the installed instrumentation in conformance with the guidance of SRP Table 6.5.1-1 and as described in Appendix 6B, for SGTS, and Appendix 9D, for CRHA HVAC systems and test connections as required by RG 1.52. Additionally, both of these systems address RG 1.52 as described in Subsection 6.5.1.3.5, Appendix 6A (compliance with RG 1.52), Subsection 9.4.1.1.7 (RG 1.52 Compliance Status), and Appendix 9C. Air filtration and adsorption units are not required for normal ventilation on ABWR, since there are no requirements for safety-related adsorption units in normal operations, except for the incinerator off-gas exhaust which is directed to a separate monitor vent (Subsection 9.4.6.5.3). Therefore, RG 1.140 is not applicable. Thus, Issue B-36 is resolved for ABWR.
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	This item was resolved in the ABWR DCD Tier 2, Section 19B.2.25, in the original design certification: All pressure containing components including all high pressure to low pressure safety-related system boundary valves used in the Advanced Boiling Water Reactor (ABWR) Standard Plant design are identified as Safety Class 1, 2, or 3, and are designed, manufactured, and tested in accordance with the guidelines of the ASME Code, Section III. (See Subsections 3.2.1, 3.2.2, and 3.2.3 for Seismic Classification, Quality Group Classifications, and Safety Classifications, respectively. Table 3.2-1 provides a cross-reference between safety and code classifications.) Boundary valves will be periodically inservice tested in accordance with the provisions of ASME Code, Section XI, to assure operational integrity as well as to Subsection IWV requirements for each valve category. Code Class 1, 2, and 3 valves will be categorized according to Subarticle IWV-2100. Valve test requirements and valve performance testing frequency are listed in the Subsections 3.9.6, 3.9.6.2, 3.9.6.2.1, 3.9.6.2.2, and

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
		3.9.6.2.3. In summary, the High Pressure and Low Pressure system boundary interface valves are designed, manufactured, pre-operational tested, and in-service tested according to the guidelines of the ASME Code and satisfy the intent of SRP Section 3.9.6, Revision 2. Therefore, Generic Safety Issue B-63 is resolved for the ABWR design.
C-10	Effective Operation of Containment Sprays in a LOCA	This item was resolved in the ABWR DCD Tier 2, Section 19B.2.28, in the original design certification: The Residual Heat Removal (RHR) system provides two independent containment spray cooling systems (on loops B and C) each having a common header in the wetwell and a common spray header in the drywell and sufficient capacity for containment depressurization by removing heat and condensing steam in both the drywell and wetwell air volumes following a LOCA. The drywell sprays also function to provide removal of fission products released during a LOCA as well as in the event of failure of the drywell head. The RHR system pumps water from the suppression pool, through the RHR heat exchangers into the wetwell and drywell spray spargers in the primary containment. The drywell spray mode is initiated by operator action post-LOCA in the presence of high drywell pressure, and is terminated by operator action. Also, drywell spray is terminated automatically as the RHR injection valve starts to open, (which results from a LOCA and reactor depressurization). The wetwell spray mode is initiated by operator action, and is terminated automatically by a LOCA or terminated by operator action. The water in the 304L stainless-steel-lined suppression pool is maintained at high purity (low corrosion attack) by the Suppression Pool Cleanup (SPCU) System. In the event of a LOCA, the SPCU function is automatically terminated to accomplish containment isolation. The pH range (5.3-8.9) is maintained to minimize any corrosive attack on the pool liner (304L SS) over the life of the plant. The post-LOCA aqueous phase pH in all areas of containment will have a flat time history (i.e., the liquid coolant will remain at its design basis pH throughout the event). The use of organic coatings within the containment

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
		has been kept to a minimum. The major use of such coatings is on the carbon steel containment liner, internal steel structures and equipment inside the drywell and wetwell. The epoxy coatings are specified to meet the requirements of Regulatory Guide 1.54 and are qualified using the standard ANSI tests, including ANSI N101.4. All safety-related equipment in the containment is environmentally qualified, and protected against spray actuation (Section 3.11). The system design adheres to the appropriate criteria guidelines of ANSI/ANS 56.5-1979. Application of accepted human factors principles and methodologies to the RHR System instrumentation and controls design minimizes the possibility of inadvertent actuation as a result of operator error (Subsection 18.3.1). Pre-operational testing for operability is performed on the RHR Containment Spray Subsystem (Subsection 14.2.12.1.8). Technical Specifications/Limiting Conditions for Operation (LCOs) of the RHR Containment Spray Subsystem and the Primary Containment System are given in Chapter 16, Section 3.6. It should be noted that credit is not taken for any fission product removal provided by the drywell and wetwell spray portions of the RHR system. The quantity of fission products released into the environment following postulated accidents is controlled by the standby gas treatment system (SGTS) that has the redundancy and capability to filter the gaseous effluent from the primary and the secondary containment. The ABWR Design fulfills the requirements of General Design Criteria 41, 42, and 43 relating to fission product removal, periodic inspection, and functional testing by conforming to the criteria guidelines of SRP Section 6.5.2, Revision 2 (Subsections 3.1.2.4.12.2, 3.1.2.4.13.2, and 3.1.2.4.14.2). In summary, the ABWR design meets the intent of the criteria guidelines of SRP Section 6.5.2, Revision 2, and BTP MTEB 6-1 in order to fulfill the function of reducing the concentration of radioactive iodine and particulates in the containment atmosphere during and after a LOCA, while also minimizing the probability of initiating stress corrosion cracking of stainless steel in the safeguard systems. Design features also minimize the probability of inadvertent actuation of the RHR Containment Spray subsystem or the SGTS, thus minimizing possible damage to safety-related equipment in the containment.

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
		Technical Specifications/LCOs are also provided. Issue C-10 in NUREG-0933 is, therefore, resolved for the ABWR Standard Plant design.
GSI-75	Generic Implications of ATWS Events at the Salem Nuclear Plant	This item was resolved in the ABWR DCD Tier 2, Section 19B.2.38, in the original design certification: The reactor protection (trip) system (RPS) design provides the capability for the ABWR to satisfy the NRC requirements indicated in Generic Letter 83-28 and in NUREG-1000. Execution of the programs in the Acceptance Criteria fall primarily into the phase of operations and maintenance that are the responsibility of the COL applicant. However, Section 3.2 provides the safety-related classification of principal components for the second criterion of the Acceptance Criteria. Therefore, this issue, 75, is resolved for ABWR.
GSI-86	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	This item was resolved in the ABWR DCD Tier 2, Section 19B.2.41, in the original design certification: For the ABWR, IGSCC resistance is achieved through the use of Type 316 stainless steel and compliance with the guidelines of NUREG-0313. All materials are supplied in the solution heat treated condition. During fabrication, any heating operations (except welding) between 800 K (427°C) and 1255 K (982°C) are avoided, unless followed by solution heat treatment. The ABWR water is maintained at the lowest practically achievable impurity levels to minimize its corrosion potential. In summary, only stainless steel Type 316 material is used and the piping is fabricated, tested and installed in accordance with ASME Code, Section III, (Reference 19B.2.41-3) and NUREG-0313. Also, the owner-operator is required to comply with ASME Code, Section XI, (Reference 19B.2.41-3) for the performance of inservice inspection. Therefore, this issue is resolved for the ABWR Standard Plant design.
GSI-89	Stiff Pipe Clamps	This item was resolved in the ABWR DCD Tier 2, Section 19B.2.43, in the original design certification:

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
		For the ABWR, the following stiff pipe clamp parameters will be very similar to those for the BWR stiff pipe clamps evaluated in the calculations summarized above: • Stiff pipe clamp geometry and material properties • Pipe schedule and material properties • Support rated loads less than or equal to 2.26E+05 N • Piping system operating pressures and temperatures and operating transients • Piping stresses at branch connections and elbows much greater than at stiff clamp locations Therefore, it can be concluded that the governing ABWR piping stresses will not occur at stiff pipe clamp locations. For the ABWR, the piping design specifications shall require that stiff pipe clamps be installed on straight runs of pipe or on bends with a radius of at least five pipe diameters. The pipe clamp induced stresses for NSS piping can then be considered negligible and do not warrant explicit consideration. The piping design specifications shall require that if stiff clamps are used on other than NSS piping, the stresses they induce will be considered. This issue is resolved for the ABWR.
GSI-124	Auxiliary Feedwater System Reliability	This issue is not applicable to BWRs and, therefore, is resolved for the ABWR (see ABWR DCD Tier 2, Section 19B.2.51, in the original design certification).
GSI-163	Multiple Steam Generator Tube Leakage	This is a PWR issue and is not applicable to ABWR.
GSI-186	Potential risk and consequences of heavy load drops	This item was resolved in the ABWR DCD Tier 2, Section 9.1.5.5, in the original design certification: 9.1.5.5 Safety Evaluations The cranes, hoists, and related lifting devices used for handling heavy loads either satisfy the single-failure guidelines of NUREG-0612, Subsection 5.1.6, including

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
		NUREG-0554 or evaluations are made to demonstrate compliance with the recommended guidelines of Section 5.1, including Subsections 5.1.4 and 5.1.5. The equipment handling components over the fuel pool are designed to meet the single-failure-proof criteria to satisfy NUREG-0554. Redundant safety interlocks and limit switches are provided to prevent transporting heavy loads other than spent fuel by the refueling bridge crane over any spent fuel that is stored in the spent-fuel storage pool. A transportation routing study will be made of all planned heavy load handling moves to evaluate and minimize safety risks. Safety evaluations of related light loads and refueling handling tasks in which heavy load equipment is also used are covered in Subsection 9.1.4.3. The CRD and RIP maintenance equipment on the rotating bridge below the RPV used during refueling operation will be withdrawn through the personnel equipment tunnel to outside primary containment.
GSI-189	Susceptibility of Ice Condenser containments to early failure from hydrogen combustion during a severe accident (includes Mark III containments)	This item was resolved in the ABWR DCD Tier 2, Section 19B.2.18A-48, in the original design certification: The ABWR containment is inerted and per 10 CFR 50.34 (f)(2)(ix) can withstand the pressure and energy addition from a 100% fuel-clad metal-water reaction. However, in the ABWR, there are no design-basis events that result in core uncovery or core heatup sufficient to cause significant metal-water reaction. Section 6.2.5.3 states that this is equivalent to the reaction of the active clad to a depth of 5.842E-3 mm (0.00023 inches) or 0.72% of the active clad. Therefore, this issue is resolved for the ABWR.
GSI-191	Assessment of Debris Accumulation on PWR Sump Performance	This is a PWR issue and is not applicable to ABWR.
GSI-193	BWR ECCS Suction Concerns (gas intrusion)	DCD, Tier 2, Section 19B.2.2 A-1, addresses Water Hammer. To the extent that additional action is deemed necessary, this issue would be addressed in a COLA or in a future amendment of the design certification. No changes to the ABWR certified design are included in the renewal application to address GSI-193.

NUREG-0933, Appendix B, Number	Description	Action/Status of Resolution
GSI-199	Implications of updated seismic hazard estimates in central and eastern U.S.	To the extent that any COLA may reference the ABWR certified design, the applicant could determine if GSI-199 needs to be addressed. GEH may consider amending the ABWR design certification in the future, but no changes to the ABWR certified design are included in the renewal application to address GSI-199.

Enclosure 2

MFN 10-342

CD – ABWR DCD, Revision 5, Introduction

Enclosure 3

MFN 10-342

CD – ABWR DCD, Revision 5, Tier 1 and Tier 2
With Security-Related Information

Security-Related Information Notice

Enclosure 3 contains Security-Related Information identified by the designation
"{{{Security-Related Information – Withhold Under 10 CFR 2.390}}}."
GEH hereby requests that this information be withheld from
public disclosure in accordance with the provisions of 10 CFR 2.390.

Enclosure 4

MFN 10-342

CD – ABWR DCD, Revision 5, Tier 1 and Tier 2
Public Version
Without Security-Related Information

Enclosure 5

MFN 10-342

SAMDA Technical Support Document for the ABWR
Revision 1

GE Nuclear Energy

December 21, 1994

MFN No. 162-94
Docket No. 52-001

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: R. W., Borchardt, Director
Standardization Project Directorate

Subject: **NEPA/SAMDA Submittal for the ABWR**

Reference:
1. Letter, J.F. Quirk to R.W. Borchardt, same title, August 26, 1993, MFN No. 137-93
2. Letter, J.F. Quirk to R.W Borchardt, same title, November 18, 1994, MFN No. 148-94

The attached Technical Support Document (TSD) for the ABWR supersedes the TSD transmitted August 26, 1993 (Reference 1) and November 18, 1994 (Reference 2). On December 15, 1994, GE discussed the staff's comments on Reference 2. This updated version of the TSD incorporates staff comments.

The conclusions regarding radiological risk from severe accidents in plants of ABWR design remain unchanged and GE believes that this TSD provides a sufficient basis for the NRC to issue proposed amendments to 10CFR Part 52 which concludes:

1) for the ABWR design, all reasonable steps have been taken to reduce the occurrence of a severe accident involving substantial damage to the core and to mitigate the consequences of such an accident should one occur;

2) no cost-effective SAMDAs to the ABWR design have been identified to prevent or mitigate the consequences of a severe accident involving substantial damage to the core; and,

3) no further evaluation of severe accidents for the ABWR design, including SAMDAs to the design, is required in any environmental report, environmental assessment, environmental impact statement or other environmental analysis prepared in connection with issuance of a combined license for a nuclear power plant referencing a certified ABWR design.

2222 See Attached distribution

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If you have any questions on the attached TSD, please call Peter D. Knecht at (408) 925-6215.

Sincerely,

Joseph F. Quirk
Project Manager
ABWR Certification
MC-782 (408) 925-6219

Att.

cc:	S.A. Hucik	(GE)	N.D. Fletcher	(DOE)
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ıECHNICAL SUPPORT DOCUMENT FOR THE ABWR

General Electric Company
San Jose, California
December 1994

TECHNICAL SUPPORT DOCUMENT FOR THE ABWR

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EXECUTIVE SUMMARY

The term "severe accident" refers to those events which are "beyond the substantial coverage of design basis events" and includes those for which there is substantial damage to the reactor core whether or not there are serious off-site consequences. See Severe Accident Policy Statement, 50 Fed. Reg. 32,138 and 32,139 (August 8, 1985).

For new reactor designs, such as the ABWR, the Nuclear Regulatory Commission (NRC), in satisfaction of its severe accident safety requirements and guidance, is requiring, among other things, the evaluation of design alternatives to reduce the radiological risk from a severe accident by preventing substantial core damage (i.e., preventing a severe accident) or by limiting releases from the containment in the event that substantial core damage occurs (i.e., mitigating the impacts of a severe accident).

The National Environmental Policy Act (NEPA) requires the consideration of reasonable alternatives to proposed major Federal actions significantly affecting the quality of the human environment, including alternatives to mitigate the impacts of the proposed action. In 1989, a Federal Court of Appeals determined that NEPA required consideration of certain design alternatives; namely, severe accident mitigation design alternatives (SAMDAs). See Limerick E... Action v. NRC, 869 F.2d 719 (3rd Cir. 1989). The court indicated that "[SAMDAs] are, as the name suggests, possible plant design modifications that are intended not to prevent an accident, but to lessen the severity of the impact of an accident should one occur." Id. at 731. The court rejected the use of a policy statement as an acceptable basis for closing out NEPA consideration of SAMDAs in a licensing proceeding, because, among other things, it was not a rule making. Id. at 739.

Recently, the NRC Staff expanded the concept of SAMDAs to encompass design alternatives to prevent severe accidents, as well as mitigate them. See NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," (Volume I, p. 5-100). By doing so, the Staff makes the set of SAMDAs considered under NEPA the same as the set of alternatives to prevent or mitigate severe accidents considered in satisfaction of the Commission's severe accident requirements and policy.

This document provides the technical basis for determining the status of severe accident closure under NEPA for the ABWR design. The report concludes that there is an adequate technical basis for closure of severe accidents under NEPA for the ABWR design. The basis and conclusions are expected to be codified in the form of proposed amendments to 10 CFR Part 52. The amendments would provide that:

(1) For the ABWR design, all reasonable steps have been taken to reduce the occurrence of a severe accident involving substantial damage to the core and to mitigate the consequences of such an accident should one occur;

(2) No cost-effective SAMDAs to the ABWR design have been identified to prevent or mitigate the consequences of a severe accident involving substantial damage to the core;

(3) No further evaluation of severe accidents for the ABWR design, including SAMDAs to the design, is required in any environmental report, environmental assessment, environmental impact statement or other environmental analysis prepared in connection with issuance of a combined license for a nuclear power plant referencing a certified ABWR design; and,

1.0 INTRODUCTION

1.1 Background

The term "severe accident" refers to those events that are "beyond the substantial coverage of design basis events" and includes those for which there is substantial damage to the reactor core whether or not there are serious off-site consequences. See Severe Accident Policy Statement, 50 Fed. Reg. 32,138 and 32,139 (August 8, 1985). For new reactor designs, such as the ABWR, the Nuclear Regulatory Commission (NRC), in satisfaction of its severe accident safety requirements, is requiring, among other things, the evaluation of design alternatives to reduce the radiological risk from a severe accident by preventing substantial core damage (i.e., preventing a severe accident) or by limiting releases from the containment in the event that substantial core damage occurs (i.e., mitigating the impacts of a severe accident).

The Commission's severe accident safety requirements for new designs are set forth in 10 CFR Part 52, §52.47(a)(1)(ii), (iv) and (v). Paragraph 52.47(a)(1)(ii) references the Commission's Three Mile Island safety requirements in §50.34(f). Paragraph 52.47(a)(1)(iv) concerns the treatment of unresolved safety issues and generic safety issues. Paragraph 52.47(a)(1)(v) requires the performance of a design-specific probabilistic risk assessment (PRA). The Commission's Severe Accident Policy Statement elaborates what the Commission is requiring for new designs. The Commission's Safety Goal Policy Statement (51 Fed. Reg. 30,028 (August 21, 1986)) sets goals and objectives for determining an acceptable level of radiological risk.

As part of its application for certification of the ABWR design, GE has prepared a Standard Safety Analysis Report (ABWR SSAR). Chapter 19 of the ABWR SSAR, "Response to Severe Accident Policy Statement," demonstrates how the ABWR design meets the Commission's severe accident safety requirements and policies. In particular, Chapter 19 includes:

(1) Identification of the dominant severe accident sequences and associated source terms for the ABWR design;

(2) Descriptions of modifications that have been made to the ABWR design, based on the results of the Probabilistic Risk Assessment (PRA), to prevent or mitigate severe accidents and reduce the risk of a severe accident;

(3) Bases for concluding that "all reasonable steps [have been taken] to reduce the chances of occurrence of a severe accident involving substantial damage to the reactor core and to mitigate the consequences of such an accident should one occur," (Severe Accident Policy Statement (50 Fed. Reg. 32,139)); and

(4) Bases for concluding that the ABWR meets Commission's Safety Goals and objectives as set forth in the Safety Goal Policy Statement

Conse ly, the conclusions are drawn in Chapter 19 that further modifications to the ABWR design educe severe accident risk are not warranted. The National Environmental Policy Act (NEPA) requires the consideration of reasonable alternatives to proposed major Federal actions significantly affecting the quality of the human environment, including alternatives to mitigate the impacts of the proposed action. In 1989, a Federal Court of Appeals determined that NEPA required consideration of certain design alternatives; namely, severe accident mitigation design alternatives (SAMDAs). Limerick Ecology Action v. NRC, 869 F.2d 719 (3rd Cir. 1989). The court indicated that "[SAMDAs] are, as the name suggests, possible plant design modifications that are intended not to prevent an accident, but to lessen the severity of the impact of an accident should one occur." Id. at 731. The court rejected the use of a policy statement as an acceptable basis for closing out NEPA consideration of SAMDAs in a licensing proceeding, because, among other things, it was not a rule making, see id. at 739.

Subsequent to the Limerick decision, the NRC issued Supplemental Final Environmental Impact Statements for the Limerick and Comanche Peak facilities that considered whether there were any cost-effective SAMDAs that should be added to these facilities ("NEPA/SAMDA FES Supplements"). On the basis of the evaluations in the supplements (called "NEPA/SAMDA evaluations"), the NRC determined that further modifications would not be cost-effective and were not necessary in order to satisfy the mandates of NEPA.

In recognition of the Limerick decision, the Commission is requiring NEPA consideration in Part 52 licensing of whether there are cost-effective SAMDAs that should be added to a new reactor design to reduce severe accident risk. While this consideration could be done later on a facility-specific basis for each combined license application under Subpart C to Part 52, the Commission has decided that maintenance of design standardization will be enhanced if this is done on a generic basis for each standard design in conjunction with design certification. See SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs." That is, the Commission has decided to resolve the NEPA/SAMDA question through rule-making at the time of certification in a so called unitary proceeding, rather than in the context of later licensing proceedings

Recently, the NRC Staff expanded the definition of SAMDAs to encompass design alternatives to prevent severe accidents, as well as mitigate them. See NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," (Volume I, p. 5-100). By doing so, the Staff makes the set of SAMDAs considered under NEPA the same as the set of alternatives to prevent or mitigate severe accidents considered in satisfaction of the Commission's severe accident requirements and policies.

1.2 Purpose

The purpose of this technical support document is to provide a basis for determining the status of severe accident closure under NEPA for the ABWR design. The document supports a determination, which could be codified in a manner similar to the format of the Waste

Confidence Rule (10 CFR §51.23), as proposed in amendments to 10 CFR Part 52. These amendments would provide that:

(1) For the ABWR design, all reasonable steps have been taken to reduce the occurrence of a severe accident involving substantial damage to the core and to mitigate the consequences of such an accident should one occur;

(2) No cost-effective SAMDAs to the ABWR design have been identified to prevent or mitigate the consequences of a severe accident involving substantial damage to the core;

(3) No further evaluation of severe accidents for the ABWR design, including SAMDAs to the design, is required in any environmental report, environmental assessment, environmental impact statement or other environmental analysis prepared in connection with issuance of a combined license for a nuclear power plant referencing a certified ABWR design; and,

The evaluation presented in this document is modeled after that found in the Limerick and Comanche Peak NEPA/SAMDA FES Supplements for those facilities. Additional information concerning the radiological risk from severe accidents for those plants is not found in the supplements, but in the FESs for the Limerick and Comanche Peak facilities. That information with respect to the ABWR design is presented in this document. The discussion herein of the radiological risk from severe accidents is based on Chapter 19 of the ABWR SSAR. Attachment A to this document presents the basis for concluding that further modifications to the ABWR design are not warranted in order to reduce the risk of a severe accident through the addition of design features to prevent or mitigate a severe accident. This information originally appeared as Appendix P to Chapter 19 of the SSAR. It was subsequently agreed with the NRC staff that this information should be set forth in an attachment to this document; accordingly, it has been located, in updated form, as Attachment A hereto.

1.3 Description of Technical Support Document

Section 2.0 provides an overview of the radiological risks from severe accidents. Sections 3.0 through 5.0 provide the NEPA/SAMDA analysis. Section 3.0 discusses the methodological approach to the evaluation of SAMDAs under NEPA. Section 4.0 presents the results of the cost-effectiveness evaluation of the potential SAMDA modifications. Section 5.0 presents the conclusions and Section 6.0 the references.

2.0 EVALUATIONS OF RADIOLOGICAL RISK FROM NUCLEAR POWER PLANTS

2.1 Evaluation of SAMDAs Under NEPA and Limerick Ecology Action

Limerick Ecology Action stands for two propositions. First, NEPA requires explicit consideration of SAMDAs unless the Commission makes a finding that the severe accidents being mitigated are remote and speculative. Second, the Commission may not make this finding and dispose of

NEPA consideration of SAMDAs by means of a policy statement. The purpose of evaluating SAMDAs under NEPA is to assure that all reasonable means have been considered to mitigate the impacts of severe accidents that are not remote and speculative. As discussed above, the Commission has indicated that it will resolve the NEPA/SAMDA issue for a new reactor design in the same proceeding, called a unitary proceeding, in which it certifies that design.

The Commission's Severe Accident and Safety Goal policy statements require the Commission to make certain findings about each new reactor design. For evolutionary designs, of which the ABWR is one, this must be done by the Staff in conjunction with FDA approval and by the Commission in conjunction with certification. First, the Commission must find that an evolutionary plant meets the safety goals and objectives; i.e., that the radiological risk from operating an evolutionary plant will be acceptable, meaning that any further reduction in risk will not be substantial.

Second, the Commission must find that all reasonable means have been taken to reduce severe accident risk in the evolutionary plant design. As part of the basis for making this finding, the cost-effectiveness of risk reduction alternatives of a preventive or mitigative nature must be evaluated.

Chapter 19 of the ABWR SSAR demonstrates that these findings can be made for the ABWR design. Given the nature and findings of these severe accident and safety goal evaluations, GE believes that a sufficient basis exists for finding by rule that further consideration of severe accidents, including evaluation of SAMDAs pursuant to NEPA, is neither necessary nor reasonable.

2.2 Cost/Benefit Standard for NEPA Evaluation of SAMDAs

The Limerick decision interpreted NEPA to require evaluation of SAMDAs for their risk reduction potential. In implementing the court's decision, the NRC considered the cost-effectiveness of each candidate SAMDA in mitigating the impact of a severe accident, using the $1,000 per person-rem averted standard. This standard is a surrogate for all off-site consequences.

The basic approach in this study is to rank the SAMDAs in terms of their cost-effectiveness in mitigating the impact of a severe accident. The criterion applied is the $1,000 per person-rem averted standard, which is what the Commission has historically used in distinguishing among and ranking design alternatives, including SAMDAs.

The Commission has used this standard in the context of both safety and NEPA analyses. For example, in the context of safety analysis, the standard has been used to perform evaluations associated with implementation of the Safety Goal Policy Statement; the Severe Accident Policy Statement; and §50.34(f) requirements. In the context of environmental analysis, it has been used in the Limerick and Comanche Peak NEPA/SAMDA FES Supplements; and in the draft Generic Environmental Impact Statement for License Renewal of Nuclear Plants (NUREG-1437).

As indicated above, the Commission is preparing a Generic Environmental Impact Statement for License Renewal of Nuclear Plants. The draft statement, NUREG-1437, makes clear that the use of this standard in the evaluation of severe accident risk reduction alternatives, which include SAMDAs, is acceptable (see NUREG-1437, Vol. I, p. 5-108).

On the basis of these considerations, the cost/benefit ratio of $1,000 per person-rem averted is viewed as an acceptable standard for the purposes of evaluating SAMDAs under NEPA.

2.3 Socio-Economic Risks for Severe Accidents

As discussed above in Section 2.2, the Commission uses the $1,000/person-rem-averted standard as a surrogate for all off-site consequences. See SECY-89-102, "Implementation of Safety Goal Policy." However, Environmental Impact Statements (EIS) for nuclear power plants provide separate, general discussions of the socio-economic risks from severe accidents. In keeping with this precedent, GE is providing a general discussion of socio-economic risks for the ABWR design, based in large measure on the discussion of such risks in NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants."

The term "socio-economic risk from a severe accident" means the probability of a severe accident multiplied by the socio-economic impacts of a severe accident. "Socio economic impacts," in turn, relate to off-site costs. The off-site costs considered in NUREG-1437 (see Vol. I, p. 5-90) are:

- Evacuation costs
- Value of crops or milk, contaminated and condemned
- Costs of decontaminating property where practical
- Indirect costs due to the loss of the use of property or incomes derived therefrom (including interdiction to prevent human injury), and
- Impacts in wider regional markets and on sources of supply outside the contaminated area.

NUREG-1437 estimated the socio-economic risks from severe accidents. The estimates were based on 27 FESs for nuclear power plants that contain analyses considering the probabilities and consequences of severe accidents. For these plants, the off-site costs were estimated to be as high as $6 billion to $8 billion dollars for severe accidents with a probability of once in one million operating years of occurring. Higher costs were estimated for severe accidents with much lower probabilities. The projected cost of adverse health effects from deaths and illnesses were estimated to average about 10-20% of off-site mitigation costs and were not included in the $6-$8 billion dollar estimate.

Another source of costs, which NUREG-1437 indicated could reach into the billions of dollars, was costs associated with the termination of economic activities in a contaminated area, which would create adverse economic impacts in wider regional markets and sources of supplies outside the contaminated area. The predicted conditional land contamination was estimated to be small (10 acres/year at most). (See NUREG-1437, Vol. I, pp. 5-90 through 5-93.)

NUREG-1437 provides the bases for concluding that the socio-economic risks from severe accidents are predicted to be small and the residual impacts of severe accidents so minor that detailed consideration of mitigation alternatives is not warranted. See 56 Fed. Reg. 47,016, 47,019, 47,034 and 47,035 (September 17, 1991).

The socio-economic risks contained in NUREG-1437 are bounding for plants of ABWR design. First, the core damage frequency for plants of ABWR design is 1.6E-7 per year. Thus, no accidents, and hence no off-site costs, are expected at probabilities at or greater than once in one million years. Second, plants of ABWR design meet the safety goals set forth by the NRC. See Section 3.2, below.

3.0 RADIOLOGICAL RISK FROM SEVERE ACCIDENTS IN PLANTS OF ABWR DESIGN

3.1 Severe Accidents in Plants of ABWR Design

Chapter 19 of the ABWR SSAR, "Response to Severe Accident Policy Statement," establishes that the Commission's severe accident safety requirements have been met for the ABWR design, including treatment of internal and external events, uncertainties, performance of sensitivity studies, and support of conclusions by appropriate deterministic analyses and the evaluations required by 10 CFR Part 50.34(f). It also establishes that the Commission's safety goals have been met.

Specifically, the following topics were addressed in Chapter 19 of the ABWR SSAR:

(1) Consideration of the contributions of internal events (Section 19.3), Shutdown events (Section 19.4) and external events (Section 19.4) to severe accident risks, including a seismic risk analysis based on the application of the seismic margins methodology (Appendix 19I);

(2) Identification of the ABWR dominant accident sequences;

(3) Identification of severe accident risk reduction features which were included in the ABWR design to achieve accident prevention and mitigation (addressed in Subsection 19.7.3(2));

Consideration of additional modifications, evaluated in accordance with §50.34(f)(1), is addressed in Attachment A. Chapter 19 concludes that the severe accident requirements of 10 CFR Part 52 (§52.47 (a)(1)(ii), (iv) & (v)) and the Severe Accident Policy Statement have been met. It also provides a summary of the bases for these conclusions. In particular, Chapter 19 presents a summary of the bases for concluding that the requirements of § 50.34(f) (referenced in §52.47(a)(1)(ii)) have been met, including §50.34(f)(1)(i), which requires "perform[ance of] a plant/site-specific [PRA], the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant." Attachment A presents the bases for concluding that further modifications to the ABWR design are not warranted in order to reduce the risk of a severe accident through the addition of design features to prevent or mitigate a severe accident.

Section 19.6 of the ABWR SSAR addresses how the goals of the Severe Accident Policy Statement have been met for plants of ABWR design. These goals include:

- Prevention of core damage
- Prevention of early containment failure for dominant accident sequences
- Evaluation of the effects of hydrogen generation
- Heat removal to reduce the probability of containment failure
- Prevention of hydrogen deflagration and detonation
- Offsite dose, and
- Containment conditional failure probability.

Specific conclusions concerning severe accidents for plants of ABWR design based on the ABWR SSAR Chapter 19 evaluations are as follows:

(1) Core Damage Frequency. The ABWR core damage frequency was determined to be 1.6E-7 per reactor year in Subsection 19.6.2. The goal was 1E-6 per reactor year.

(2) Conditional Containment Failure Probability. The conditional containment failure probability was shown to be 0.002 in Subsection 19.6.8. This is significantly below the goal of 0.1.

(3) Individual Risk (Prompt Fatality Risk). The prompt fatality risk to a biologically average individual within one mile of an ABWR site boundary was determined to be 1.4E-13 per individual per year in Section 19E.3. This is significantly less than the goal of one tenth of one percent of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. Population are generally exposed. The numerical value of this goal is 3.9E-7 per individual per year (or 0.04 per 100,000 people per year).

(4) Societal Risk (Latent Fatality Risk). The latent fatality risk to the population within 50 miles of an ABWR site boundary was determined to be 9.0E-13 per individual per year in Section 19E.3. This is significantly less than the goal of one tenth of one percent of the sum of the cancer fatality risks resulting from all other causes. The numerical value of this goal is 1.7E-6 per individual per year (or 0.17 deaths per 100,000 people per year).

(5) Probability of Large Off-Site Dose. The probability of exceeding a whole body dose of 25 rem at a distance of one-half mile from a ABWR was determined to be less than 1E-9 per reactor year in Section 19E.3.

Residual radiological risk from severe accidents in plants of ABWR design is summarized in Table A-1 (reproduced here as Table 1). The cumulative exposure risk to the population within 50 miles of a plant of ABWR design is approximately 0.269 person-rem for an assumed plant life of 60 years. This calculation includes the dominant sequences, as well as several sequences that are considered remote and speculative.

3.2 Dominant Severe Accident Sequences for Plants of ABWR Design

In performing the PRA for the ABWR design, GE identified and evaluated many severe accident sequences. For each sequence, the analysis identified an initiating event and traced the accident's progression to its end. For sequences involving core damage, conditional containment failure probabilities and offsite consequences were estimated. After the accident scenarios were binned according to radiological release (source term) parameters, only two dominant cases remained.

The dominant cases are: Case 1 (best estimate core damage sequences that had rupture disk activation); and the NCL case (core damage with normal containment leakage). The residual risks of these two cases can be found in Table 1. The complete radiological consequence analysis of the dominant sequences can be found in Section 19E.3 of the ABWR SSAR.

The probability of occurrence of dominant sequences is greater than 1E-9 per year. Several sequences with occurrence probabilities less than 1E-9 per year were carried through the severe accident analysis in order to determine the sensitivity of plants of ABWR design to certain phenomena and parameters. These sequences were also considered in the SAMDA evaluation for sensitivity purposes.

Sequences with probabilities of occurrence less than 1E-9 were considered remote and speculative. While the Commission has not yet specified a quantitative point at which it will consider severe accident probabilities as remote and speculative, it has indicated that a decision to consider severe accidents remote and speculative would be based upon the accident probabilities and the accident scenarios being analyzed. See Vermont Yankee Nuclear Power Corporation, (Vermont Yankee Nuclear Power Station), CLI-90-07, 32 NRC 129, 132 (1990).

GE believes that the severe accident analysis in Chapter 19 of the ABWR SSAR provides a sufficient basis for the Commission to find that ABWR sequences that are not dominant can be deemed remote and speculative.

3.3 Overall Conclusions from Chapter 19 of the ABWR SSAR

The specific conclusions about severe accident risk discussed above support the overall conclusion that the environmental impacts of severe accidents for plants of ABWR design represent a low risk to the population and to the environment. For the ABWR design, all reasonable steps have been taken to reduce the occurrence of a severe accident involving substantial damage to the core and to mitigate the consequences of such an accident should one occur. No further cost-effective modifications to the ABWR design have been identified to reduce the risk from a severe accident involving substantial damage to the core. No further evaluation of severe accidents for the ABWR design is required to demonstrate compliance with the Commission's severe accident requirements or policy or the safety goal.

4.0 COST/BENEFIT EVALUATION OF SAMDAS FOR PLANTS OF ABWR DESIGN

4.1 SAMDA Definition Applied to Plants of ABWR Design

Attachment A considers whether the ABWR design should be modified in order to prevent or mitigate the consequences of a severe accident in satisfaction of the NRC's severe accident requirements in 10 CFR Parts 50 & 52 and the Severe Accident Policy Statement. The cost/benefit evaluation of SAMDAs to plants of ABWR design uses the expanded definition of SAMDAs set forth in NUREG-1437: design alternatives that could prevent and/or mitigate the consequences of a severe accident.

4.2 Cost/Benefit Standard for Evaluation of ABWR SAMDAs

As discussed in Section 2.2 above, the cost/benefit ratio of $1,000 per person-rem averted is viewed by the NRC and the nuclear industry as an acceptable standard for the purposes of evaluating SAMDAs under NEPA. This standard was used as a surrogate for all off-site costs in the cost/benefit evaluation of SAMDAs to plants of ABWR design. Averted on-site costs were incorporated for SAMDAs that were at least partially preventive in nature[1]. On-site costs resulting from a severe accident include replacement power, on-site cleanup costs, and economic loss of the facility. A more detailed discussion of averted on-site costs can be found in Attachment A. The equation used to determine the cost/benefit ratio is:

$$\text{Cost/benefit ratio} = \frac{\text{Cost of SAMDA implementation MINUS averted on-site costs}}{\text{Reduction in residual risk (person-rem/plant life)}}$$

A plant lifetime of 60 years was assumed to maximize the reduction in residual risk.

[1]Assessment of averted on-site costs are provided for information only. It is GE's position that the NRC is not required to account for these costs.

4.3 Candidate SAMDAs for the ABWR Design

The complete list of SAMDAs considered for plants of ABWR design is contained in Table 2. This list is also contained in Table A-3 of Attachment A. The SAMDAs are classified according to the following categories:

(1) Modification is applicable to the ABWR and already incorporated into the design. No further evaluation is needed.

(2) Modification is applicable to the ABWR but not incorporated into the design. These modifications were considered further in Attachment A and the results of the cost/benefit analysis will be presented in this document.

(3) Modification is not applicable to the ABWR design due to the basis provided.

(4) Modification is considered as part of another modification listed in the table.

Table 3 lists the advantages and disadvantages of each design alternative that is applicable to the ABWR but not incorporated into the design ("2" classification in Table 2). A detailed discussion of each alternative is contained in Section A.4 of Attachment A.

4.4 Cost Estimates of Potential Modifications to the ABWR Design

Table 4 provides a brief explanation of the estimated costs of each design alternative applicable to the ABWR design. Details of the cost estimation methodology are provided in Section A.1.3.2 of Attachment A. As discussed in Attachment A, rough order of magnitude costs, biased in favor of making a modification, were assigned to each modification. The costs represent the incremental costs that would be incurred in a new plant rather than costs that would apply on a backfit basis.

The estimated costs of design alternatives that are, at least partially, preventive in nature were adjusted for averted on-site costs. This adjustment is included in the cost estimates in Table 4. Design alternatives that are purely mitigative in nature are not assigned any averted on-site costs because these modifications do not significantly affect site clean up cost nor significantly lessen the plant investment loss. Section A.5 of Attachment A discusses the bases for assigning averted on-site costs in detail.

Considerable uncertainties prevent precise cost estimates because design details have not been developed and construction and licensing delays cannot be accurately evaluated. For purpose of this evaluation, all known or reasonably expected costs were accounted for in order that a reasonable assessment of the minimum cost could be obtained. Using a minimum cost favors implementation of a modification. Actual implementation costs are expected to be significantly higher than those used in this evaluation.

4.5 Benefits of Potential Modifications to the ABWR Design

Table 5 summarizes the basis for assigning a benefit to each SAMDA. In general, benefits were estimated from the PRA results of Chapter 19 of the ABWR SSAR by considering which sequences are affected by each modification. Detailed discussion of the method for estimating benefit is provided in Section A.4 of Attachment A. The averted residual risk for each SAMDA is also given in Table 5.

4.6 Cost/Benefit Comparison of SAMDAs

Table 6 summarizes the results of combining the cost estimates from Table 4 with the benefit estimates from Table 5. As is evident from Table 6, none of the SAMDAs requires further evaluation since the cost/benefit standard was not met. The closest design alternative exceeds the criteria by more than a factor of 1000.

On the basis of the small residual risk of a plant of ABWR design, 0.269 person-rem for the entire plant life, a design modification would have to cost $269 or less in order to meet the standard of $1,000 per person-rem averted.

5.0 SUMMARY AND CONCLUSIONS

A reasonable ar ' comprehensive set of candidate SAMDAs relevant to the ABWR design was evaluated in terr of minimum costs, averted on-site costs and potential benefits. A screening criterion of $1,000 per person-rem averted was used to determine which alternatives, if any, were cost-effective. None was found to meet the criterion. In fact, the implementation cost of a SAMDA would have to be less than $269 in order to pass. Given the low residual risk profile of the ABWR design, SAMDAs cannot be reasonably incorporated in a cost-effective manner.

On the basis of the foregoing analysis, further incorporation of SAMDAs into the ABWR design is not warranted. No further screening of SAMDAs is needed and no SAMDAs need be incorporated into ABWR design in satisfaction of NEPA.

6.0 REFERENCES

1. ABWR Standard Safety Analysis Report, 23A6100, Docket No. 52-001, GE Nuclear Energy.

2. Assessment of Severe Accident Prevention and Mitigation Features, NUREG/CR-4920, Brookhaven National Laboratory, July 1988.

3. Design and Feasibility of Accident Mitigation Systems for Light Water Reactors, NUREG/CR-4025, R&D Associates, August 1985.

4. Evaluation of Proposed Modifications to the GESSAR II Design, NEDE 30640 (Proprietary), June 1984.

5. Generic Environmental Impact Statement for License Renewal of Nuclear Plants, NUREG-1437, August 1991.

6. "Issuance of Supplement to the Final Environmental Statement-Comanche Peak Steam Electric Station, Units 1 and 2", NUREG 0775 Supplement, December 15, 1989.

7. Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, NUREG 1150, January 1991.

8. "Supplement to the Final Environmental Statement-Limerick Generating Station, Units 1 and 2", NUREG 0974 Supplement, August 16, 1989.

9. Survey of the State of the Art in Mitigation Systems, NUREG/CR-3908, R&D Associates, December 1985.

10. Technical Guidance for Siting Criteria Development, NUREG/CR-2239, Sandia National Laboratories, December 1982.

11. Title 10, Code of Federal Regulations, Part 50 and 52.

12. 50FR32138, Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, August, 1985.

13. 50FR30028, Safety Goals for the Operations of Nuclear Power Plants; Policy Statement, August 1986.

Table 1
Radiological Consequences of ABWR Accident Sequences

Case	Probability (Event/Year)*	Whole Body Exposure, 50 mile (Person-rem)	Cumulative Exposure Risk (Per-rem/60 Yr)
NCL	1.3E-07	9.60E3	0.075
1	2.1E-08	1.38E4	0.017
2	7.8E-11	8.33E3	0.00004
3	0	3.71E5	0.000
4	0	2.06E5	0.000
5	7.5E-12	9.34E4	0.00004
6	3.1E-12	2.42E6	0.004
7	3.9E-10	2.73E6	0.064
8	4.1E-10	3.20E6	0.079
9	1.7E-10	3.31E6	0.034
		Total:	0.269

* Sequences with probabilities of occurrence less than 1E-9 per year are considered remote and speculative.

Table 2
Severe Accident Mitigation Design Alternatives (SAMDAs)*
Considered for the ABWR Design

Modification	Category
1. ACCIDENT MANAGEMENT	
a. Severe Accident EPGs/AMGs	2
b. Computer Aided Instrumentation	2
c. Improved Maintenance Procedures/Manuals	2
d. Preventive Maintenance Features	4
e. Improved Accident Management Instrumentation	4
f. Remote Shutdown Station	1
g. Security System	1
h. Simulator Training for Severe Accident	4
2. REACTOR DECAY HEAT REMOVAL	
a. Passive High Pressure System	2
b. Improved Depressurization	2
c. Suppression Pool Jockey Pump	2
d. Improved High Pressure Systems	1
e. Additional Active High Pressure System	1
f. Improved Low Pressure System (Firepump)	1
g. Dedicated Suppression Pool Cooling	1
h. Safety Related Condensate Storage Tank	2
i. 16 hour Station Blackout Injection	4
j. Improved Recirculation Model	4
3. CONTAINMENT CAPABILITY	
a. Larger Volume Containment	2
b. Increased Containment Pressure Capacity	2
c. Improved Vacuum Breakers	2
d. Increased Temperature Margin for Seals	1
e. Improved Leak Detection	1
f. Suppression Pool Scrubbing	1
g. Improved Bottom Penetration Design	2

* SAMDAs include both preventive and mitigative design alternatives

Table 2 (Continued)

Modification		Category
4.	CONTAINMENT HEAT REMOVAL	
	a. Larger Volume Suppression Pool	2
	b. CUW Decay Heat Removal	1
	c. High Flow Suppression Pool Cooling	1
	d. Passive Overpressure Relief	1
5.	CONTAINMENT ATMOSPHERE MASS REMOVAL	
	a. High Flow Unfiltered Vent	3
	b. High Flow Filtered Vent	3
	c. Low Flow Vent (Filtered)	2
	d. Low Flow Vent (Unfiltered)	1
6.	COMBUSTIBLE GAS CONTROL	
	a. Post Accident Inerting System	3
	b. Hydrogen Control by Venting	3
	c. Pre-inerting	1
	d. Ignition Systems	3
	e. Fire Suppression System Inerting	3
7.	CONTAINMENT SPRAY SYSTEMS	
	a. Drywell Head Flooding	2
	b. Containment Spray Augmentation	1
8.	PREVENTION CONCEPTS	
	a. Additional Service Water Pump	2
	b. Improved Operating Response	1
	c. Diverse Injection System	4
	d. Operating Experience Feedback	1
	e. Improved MSIV/SRV Design	1
9.	AC POWER SUPPLIES	
	a. Steam Driven Turbine Generator	2
	b. Alternate Pump Power Source	2
	c. Deleted	
	d. Additional Diesel Generator	1

Table 2 (Continued)

Modification		Category
9.	(Continued)	
	e. Increased Electrical Divisions	1
	f. Improved Uninterruptable Power Supplies	1
	g. AC Bus Cross-ties	1
	h. Gas Turbine	1
	i. Dedicated RHR (bunkered) Power Supply	4
10.	DC POWER SUPPLIES	
	a. Dedicated DC Power Supply	2
	b. Additional Batteries/Divisions	4
	c. Fuel Cells	4
	d. DC Cross-ties	1
	e. Extended Station Blackout Provisions	1
11.	ATWS CAPABILITY	
	a. ATWS Sized Vent	2
	b. Improved ATWS Capability	1
12.	SEISMIC CAPABILITY	
	a. Increased Seismic Margins	1
	b. Integral Basemat	3
13.	SYSTEM SIMPLIFICATION	
	a. Reactor Building Sprays	2
	b. System Simplification	1
	c. Reduction in Reactor Bldg Flooding	1
14.	CORE RETENTION DEVICES	
	a. Flooded Rubble Bed	2
	b. Reactor Cavity Flooder	1
	c. Basaltic Cements	1

Table 3
SAMDAs Evaluated Under NEPA for the ABWR

Potential Improvement	Advantages	Disadvantages
1a. Severe Accident EPGs/AMGs	Improved arrest of core melt progress and prevention of containment failure.	None
1b. Computer Aided Instrumentation	Improved prevention of core melt sequences	Additional training
1c. Improved Maintenance Procedures/Manuals	Improved prevention of core melt sequences	Increased documentation cost
2a. Passive High Pressure System	Improved prevention of core melt sequences	High cost of additional system
2b. Improved Depressurization	Improved utilization of Low Pressure systems for prevention of core melt sequences	Cost of additional equipment
2c. Suppression Pool Jockey Pump	Improved prevention of core melt sequences	Cost of additional equipment
2d. Safety Related Condensate Storage Tank	Availability following Seismic events	Design and structural costs
3a. Larger Volume Containment (Double Free Volume)	a. Increases time before containment failure b. Increases time for recovery	a. High cost b. Containment failure not prevented c. Minor radiological benefit since risks dominated by long lived isotopes
3b. Increased Containment Pressure Capability (Sufficient pressure to withstand severe accidents)	a. Eliminates large releases	a. Extreme costs b. High temperature failures not prevented
3c. Improved Vacuum Breakers (Redundant valves in each line)	a. Reduces probability of suppression pool bypass	a. Increased maintenance and equipment costs
3d. Improved Bottom Head Penetration Design	a. Increased time for in-vessel arrest	a. Cost for equipment and analysis
4a. Larger Volume Suppression Pool (Double effective liquid volume)	a. Increases heat absorption capability within containment	a. High cost

Table 3 (Continued)

Potential Improvement	Advantages	Disadvantages
4a. (Continued)	b. Increases time for recovery of systems c. ncreases time before containment failure	b. Minor radiological benefit since risks dominated by long lived isotopes
5a. Low Flow Filtered Vent	a. Provides some scrubbing of fission products if head fails b. Reduces containment leakage if movable penetrations are degraded c. low cost	a. Probability of drywell head failure is low relative to the other containment failure modes
7a. Drywell Head Flooding (Firewater crosstie to drywell head area)	Improved prevention of core melt sequences	Additional cost of equipment
8a. Additional Service Water Pump	Improved prevention of core melt sequences	Additional cost of equipment
9a. Steam Driven Turbine Generator	Improved prevention of core melt sequences	Additional cost of equipment
9b. Alternate Pump Power Source	Improved prevention of core melt sequences	Additional cost of equipment
10a. Dedicated DC Power Supply	Additional time before containment overpressure	Marginal benefit
11a. ATWS Sized Vent	a. Provides scrubbing of fission products, except noble gases, which pass through reactor building	a. Uncertain location b. Potential for inadvertent actuation c. Floods reactor building which greatly hinders site recovery after accident d. Potential failure of electrical equipment in reactor building
13a. Reactor Building Sprays (Firewater crosstie for reactor building sprays)	Reduced release of fission products from Reactor Building	Uncertain location and unknown potential consequences from inadvertent actuation

Table 3 (Continued)

Potential Improvement	Advantages	Disadvantages
14a. Flooded Rubble Bed	Prevention of core-concrete interaction affects	Small benefit over passive flooding system.

Table 4
Cost Estimates of SAMDAs Evaluated for the ABWR Under NEPA

Potential Improvement	Cost Basis	Estimated Minimum Cost
1a. Severe Accident EPGs/AMGs	Plant specific procedure preparation beyond generic work by Owners' Group.	$ 600,000
1b. Computer Aided Instrumentation	Software modifications and interface hardw re. Credit for averted onsite cost included.	$ 599,600
1c. Improved Maintenance Procedures/Manuals	Procedure preparation. Credit for averted onsite cost included.	$ 299,000
2a. Passive High Pressure System	System hardware and installation ($1,200,000), Building modification ($550,000). Credit for averted onsite cost included.	$ 1,744,000
2b. Improved Depressurization	Logic, pneumatic supplies, piping and qualification. Credit for averted onsite cost included.	$ 598,600
2c. Suppression Pool Jockey Pump	System hardware and electrical connections. Credit for averted onsite cost included.	$ 120,000
2d. Safety Related Condensate Storage Tank	Structural analysis and material. Credit for averted onsite cost included.	$ 1,000,000
3a. Larger Volume Containment (Double Free Volume)	Double current volume at $1200/ft³. Analysis not included.	$ 8,000,000
3b. Increased Containment Pressure Capability (Sufficient pressure to withstand severe accidents)	Similar to Larger Volume Containment, but denser rebar and labor required. Assumed 50% higher cost	$ 12,000,000
3c. Improved Vacuum Breakers (Redundant valves in each line)	Eight lines at $10,000 per line	$ 100,000

Table 4 (Continued)

Potential Improvement	Cost Basis	Estimated Minimum Cost
3d. Improved Bottom Head Penetration Design	205 drives at $1,000/drive and $500,000 of analysis	$ 750,000
4a. Larger Volume Suppression Pool (Double effective liquid volume)	Assumed to be the same as Larger Volume Containment	$ 8,000,000
5a. Low Flow Filtered Vent	Hardware and Testing program	$ 3,000,000
7a. Drywell Head Flooding (Firewater crosstie to drywell head area)	Minor valve and piping modification with instrumentation	$ 100,000
8a. Additional Service Water Pump	System hardware, power supplies and support systems. Credit for averted onsite cost included.	$ 5,999,000
9a. Steam Driven Turbine Generator	System hardware, cabling and structural changes. Credit for averted onsite cost included.	$ 5,994,300
9b. Alternate Pump Power Source	400 kW generator at $300/kW. Credit for averted onsite cost included.	$ 1,194,000
10a. Dedicated DC Power Supply	5000 ft² building structure addition at $500/ft² and cabling	$ 3,000,000
11a. ATWS Sized Vent	Instrumentation and cabling in addition to training	$ 300,000
13a. Reactor Building Sprays (Firewater crosstie for reactor building sprays)	Minor valve and piping modification with instrumentation.	$ 100,000
14a. Flooded Rubble Bed	1250 ft³ of material at $1000/lb	$ 18,750,000

Table 5
Benefit Estimates of SAMDAs*
Evaluated for the ABWR Under NEPA

Potential Improvement	Benefit Basis	Averted Risk Person-REM
1a. Severe Accident EPGs/AMGs	10% improvement in mitigative actions	0.015
1b. Computer Aided Instrumentation	10% improvement in preventative actions	0.01
1c. Improved Maintenance Procedures/Manuals	10% improvement in reliability of RCIC, HPCF, RHR and LPFL	0.016
2a. Passive High Pressure System	90% reliable diverse additional high pressure system	0.069
2b. Improved Depressurization	50% reduction in manual depressurization reliability	0.042
2c. Suppression Pool Jockey Pump	10% improvement in low pressure makeup reliability.	0.002
2d. Safety Related Condensate Storage Tank	Arbitrary selection due to high suppression pool availability.	0.01
3a. Larger Volume Containment (Double Free Volume)	Elimination of drywell head failure sequences	0.15
3b. Increased Containment Pressure Capability (Sufficient pressure to withstand severe accidents)	Elimination of all cases except normal containment leakage (NCL)	0.16
3c. Improved Vacuum Breakers (Redundant valves in each line)	Elimination of Case 2 sequences	0.00004
3d. Improved Bottom Head Penetration Design	50% improvement in in-vessel arrest due to additional available time	0.057
4a. Larger Volume Suppression Pool (Double effective liquid volume)	Elimination of Case 9 sequences involving loss of suppression pool cooling systems	0.0002

* SAMDAs include both preventive and mitigative design alternatives

Table 5 (Continued)

Potential Improvement	Benefit Basis	Averted Risk Person-REM
5a. Low Flow Filtered Vent	Elimination of sequences involving initiation of containment rupture disc	0.014
7a. Drywell Head Flooding (Firewater crosstie to drywell head area)	Reduction in high temperature containment failure sequences and drywell head failure sequences	0.06
8a. Additional Service Water Pump	10% improvement in reliability of RCIC, HPCF, RHR and LPFL due to improved support systems	0.016
9a. Steam Driven Turbine Generator	Improved effective availability of EDG	0.052
9b. Alternate Pump Power Source for high pressure systems	Similar to additional high pressure system. See 2a.	0.069
10a. Dedicated DC Power Supply	Similar to additional high pressure system. See 2a.	0.069
11a. ATWS Sized Vent	Reduction in Case 9 sequences	0.03
13a. Reactor Building Sprays (Firewater crosstie for reactor building sprays)	10% reduction in consequence of sequences involving containment leakage	0.017
14a. Flooded Rubble Bed	Elimination of sequences involving core-concrete interaction.	0.001

Table 6
Comparison of Estimated Costs and Benefits on SAMDAs*
Evaluated for the ABWR Under NEPA

Potential Improvement	Estimated Minimum Cost ($)	Averted Risk Person-rem	Cost-Benefit Ratio ($K per Person-rem)
1a. Severe Accident EPGs/AMGs	$ 600,000	0.015	$ 40,000
1b. Computer Aided Instrumentation	$ 599,600	0.01	$ 59,600
1c. Improved Maintenance Procedures/Manuals	$ 299,000	0.016	$ 18,700
2a. Passive High Pressure System	$ 1,744,000	0.069	$ 25,270
2b. Improved Depressurization	$ 598,600	0.042	$ 14,250
2c. Suppression Pool Jockey Pump	$ 119,800	0.002	$ 59,900
2d. Safety Related Condensate Storage Tank	$ 1,000,000	0.01	$ 100,000
3a. Larger Volume Containment (Double Free Volume)	$ 8,000,000	0.15	$ 53,300
3b. Increased Containment Pressure Capability (Sufficient pressure to withstand severe accidents)	$ 12,000,000	0.16	$ 75,000
3c. Improved Vacuum Breakers (Redundant valves in each line)	$ 100,000	0.00004	$ 2,500,000
3d. Improved Bottom Head Penetration Design	$ 750,000	0.057	$ 13,160
4a. Larger Volume Suppression Pool (Double effective liquid volume)	$ 8,000,000	0.0002	$ 40,000,000
5a. Low Flow Filtered Vent	$ 3,000,000	0.014	$ 214,300
7a. Drywell Head Flooding (Firewater crosstie to drywell head area)	$ 100,000	0.06	$ 1,700

* SAMDAs include both preventive and mitigative design alternatives

Table 6 (Continued)

Potential Improvement	Estimated Minimum Cost ($)	Averted Risk Person-rem	Cost-Benefit Ratio ($K per Person-rem)
8a. Additional Service Water Pump	$ 5,999,000	0.016	$ 375,000
9a. Steam Driven Turbine Generator	$ 5,994,300	0.052	$ 115,300
9b. Alternate Pump Power Source	$ 1,194,000	0.069	$ 17,300
10a. Dedicated DC Power Supply	$ 3,000,000	0.069	$ 43,500
11a. ATWS Sized Vent	$ 300,000	0.03	$ 10,000
13a. Reactor Building Sprays (Firewater crosstie for reactor building sprays)	$ 100,000	0.017	$ 5,900
14a. Flooded Rubble Bed	$ 18,750,000	0.001	$ 18,750,000

ATTACHMENT A*
Evaluation of Potential Modifications to the ABWR Design

A.1 INTRODUCTION AND SUMMARY

This attachment provides a description of an evaluation of potential changes to the ABWR design in order to determine whether further modifications can be justified.

A.1.1 Background

The U.S. Nuclear Regulatory Commission's policy related to severe accidents requires, in part, that an application for a design approval comply with the requirements of 10CFR50.34(f). Item (f)(1)(i) requires performance of a plant site-specific [PRA] the aim of which is to seek improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant. Chapter 19 of the ABWR SSAR provides the base PRA of the ABWR plant.

To address this requirement, a review of potential modifications to the ABWR design, beyond those included in the Probabilistic Risk Assessment (PRA), was conducted to evaluate whether potential severe accident design features could be justified on the basis of cost per person-rem averted.

This attachment summarizes the results of GE's review and evaluation of the ABWR design. Improvements have been reviewed against conservative estimates of risk reduction based on the PRA and minimum order of magnitude costs, to determine what modifications are potentially attractive.

A.1.2 Evaluation Criteria

The benefit of a particular modification was defined to be its reduction in the risk to the general public.

Offsite factors evaluated were limited to health effects to the general public based on total exposure (in person-rem) to the population within 50 miles of the site. Five representative US regions were evaluated for selected individual ABWR sequences by the CRAC2 code. The regional results were then averaged to determine the exposures. Consistent with the standard used by the NRC to evaluate radiological impacts, health effect costs were evaluated based on a value of $1,000 per offsite person-rem averted due to the design modification.

*Attachment A is updated version of ABWR SSAR Appendix 19P of the same title.

The offsite costs for other items such as relocation of local residents, elimination of land use and decontamination of contaminated land were not considered. Reductions in the risk of incurring onsite costs including economic losses, replacement power costs and direct accident costs are considered in this evaluation as credits against in the cost of the modification.

Based on the PRA results (Section A.2), 82% of the offsite risk results from very low probability events which have high consequence. The maximum justifiable cost of a modification was determined to be $269. Therefore, based on this methodology, no modifications are justifiable. However, a variety of modifications were reviewed to establish the relative attractiveness of potential changes.

A.1.3 Methodology

The overall approach was to estimate the benefit of modifications in terms of dollar cost per total person-rem averted. Underestimated costs and overestimated benefits were assessed in order to favor modifications. Because of the uncertainties in the methodology and the desire to address severe accidents with sensible modifications, this basis is judged to be acceptable for purposes of this study.

A.1.3.1 Selection of Modifications

Potential modifications were identified from a variety of previous industry and NRC sponsored studies of preventative and mitigative features which address severe accidents. Based on this composite list of modifications considered on previous designs, potential modifications were selected for further review based on being

(1) applicable to the ABWR design, and

(2) not included in the reference PRA.

Additional detail on the selection of modifications is provided in Section A.3.

A.1.3.2 Costs Basis

Rough order of magnitude costs were assigned for each modification based on the costs of systems and system improvements determined by GE. These costs represent the estimated incremental costs that would be incurred in a new plant rather than costs that would apply on a backfit basis. Section A.5 defines the cost estimates for each of the modifications.

Even for a new plant such as the ABWR, relatively large costs (several million dollars) can be expected for some modifications if they involve modifications of the building structures or arrangement. This is because the cost of labor and material is often a function of the building area required. For other modifications which involve minor hardware addition, the cost is often dominated by the need for procedure and training additions which can amount to hundreds of thousands of dollars.

The costs estimates were intentionally biased on the low side, but all known or reasonably expected costs were accounted for in order that a reasonable assessment of the minimum cost would be obtained. Actual plant costs are expected to be higher than indicated in this evaluation. All costs are referenced to 1991 U.S. dollars. For modifications which reduce the core damage frequency, the costs of modifications (Section A.5) were further reduced by an amount proportional to the reduction present worth of the risk of averted onsite costs. Onsite costs include replacement power costs, direct accident costs (including onsite cleanup) and the economic loss of the facility. Evaluation of this credit included the following considerations:

(1) Accidents were assumed to occur at any time during the 60 year life of the plant. All onsite costs associated with the accident were evaluated as to their value at the time of the accident. The economic risk of such onsite costs was evaluated as a function of time based on the onsite costs and the core damage frequency determined by the PRA. The plant core damage frequency was considered to be constant over the life of the plant. The economic risks were then evaluated based on the present worth of the time dependent economic risks.

(2) Replacement power was based on a rate of $.013/kW-h differential as bar cost. The differential rate was assumed to be constant over the remaining life of the plant.

(3) The economic value of the facility at the time of the accident was based on a straight line depreciated value. The initial invested cost was taken at $1.4 Billion based on DOE cost guidelines.

(4) Accident costs for onsite cleanup and facility were evaluated based on escalated costs to the time of the accident. Reference accident costs to the facility were assumed to be $2 Billion.

(5) Th economic evaluations were based on a discount rate of 8% and escalation factor of 3%.

A.1.3.3 Benefit Basis

The cumulative risk of accidents occurring during the life of the plant was used as a basis for estimating the maximum benefit that could be derived from modifications. A particular modification's benefit was based on its effect on the frequency of events or associated offsite dose summarized in Tables A-1 and Table A-2. Dominant contributing failure probabilities were identified based on the PRA. Changes in these probabilities were estimated to evaluate the benefit of modifications. This basis is consistent with the approach taken in previous NRC evaluations. The cumulative offsite risk was evaluated over a 60 year plant life with no escalation in the evaluation criteria of $1,000/person-rem.

Section A.4 summarizes each concept and estimated benefit for each individual potential modification. For each modification the cost per person-rem averted was evaluated to obtain the results of the individual evaluations. These conclusions are provided in Section A.7.

A.1.4 Summary of Results

Potentially attractive modifications were selected based on previous evaluations of potential prevention and mitigation concepts applicable during severe accidents. Of the modifications applicable to the ABWR design and which were not already implemented, twenty one were selected for additional review.

None of the modifications considered met the $1,000/person-rem averted criteria. The low evaluated frequency of core damage and subsequent release of radioactive material does not support modification to the ABWR based on costs in relationship to the benefit of averted exposures.

Since the most beneficial modification was evaluated to be several orders of magnitude higher than the criteria, it was concluded that no additional modifications are warranted in the ABWR design to address severe accidents. Furthermore, due to its magnitude it can be calculated that this conclusion will not be sensitive to variations in the assumptions used in the PRA results.

A.2 SEVERE ACCIDENT RISK OF ABWR

The reference design for this study was the ABWR PRA as presented in the internal events PRA (Section 19.3 of the ABWR SSAR). This evaluation accounts for features which were included in the current ABWR design specifically to address severe accidents. These features and the reference description include:

	Design Feature	SSAR References
(1)	Firewater pump crosstie	5.4.7.1.1.10
(2)	Passive containment flooder	9.5.12
(3)	Gas turbine generator	9.5.11
(4)	Overpressure Protection	6.2.5.2.6

A summary of the core damage frequency and offsite exposure frequency with these features included is shown in Table A-1. Event frequencies used in this evaluation were the same as assumed in the base PRA. The offsite exposures shown in Table A-1 were calculated by the CRAC2 code for release cases with similar consequences. The cases can be characterized as follows:

Case 1 Core Melt arrested in vessel or in Containment with actuation of containment rupture disk.

Case 2 Low Pressure Core Melt with suppression pool bypass and actuation of containment rupture disk.

Case 3	High Pressure Core Melt with drywell Head failure and fire water spray initiation.
Case 4	Suppression Pool Decontamination reduction (Not used).
Case 5	Large Break LOCA without recovery and with actuation of containment rupture disk.
Case 6	High Pressure Core Melt with Drywell Head failure and no firewater spray initiation.
Case 7	Low Pressure Core Melt with Drywell Head failure and no mitigation
Case 8	High Pressure Core Melt with Early Containment failure.
Case 9	ATWS event with Drywell Head failure.
NCL	Normal Containment Leakage to Reactor Building.

The offsite exposures for each case shown in Table A-1 were calculated by the CRAC2 code for five representative US regions for the selected individual ABWR sequences as discussed in Section 19E.3 of the ABWR SSAR.

Table A-2 provides additional detail on the individual contributors to the total core damage frequency. As indicated on Table A-2, the core damage frequency is dominated by low pressure transient events (LCLP) (61.4%), followed by high pressure transient events (LCHP) (28.1%) and station blackout sequences (SBRC) (10.3%).

Review of Table A-1 also indicates that the dominant contributors to the ABWR offsite exposure risk are the relatively low probability (less than 4E-10/yr), high consequence events (Cases 6 through 9) which contribute about 82% of the offsite exposure risk.

A.3 POTENTIAL ABWR MODIFICATIONS

Potential modifications to the ABWR design were derived from a survey of various studies indicated in References A-1 through A-7 and the ABWR design process discussed in Section 19.7 of the ABWR SSAR. From these, a composite list of modifications was established. This list of potential modifications was reviewed to identify concepts which were already included in the ABWR design or which are not applicable.

Table A-3 summarizes the complete list of modifications and their classification according to the following categories:

(1) Modification is applicable to ABWR and already incorporated in the ABWR design. No further evaluation is needed.

(2) Modification is applicable to ABWR and not incorporated in ABWR design. (Table A-4 lists the Category 2 modifications which are evaluated further in this attachment.)

(3) Modification is not applicable to the ABWR design due to the basis provided.

(4) Modification is applicable to ABWR and is incorporated with the referenced modification.

A.4 RISK REDUCTION OF POTENTIAL MODIFICATIONS

This section provides evaluations of the benefits of potential modifications to the ABWR design identified in Table A-4. For each modification the basis for the evalu..ion and the concept is described. Table A-5 summarizes the benefit in terms of person-rem averted risk for each of the evaluated modifications.

A.4.1 Accident Management

Accident management is a current topic under generic development within the Industry through the development of Accident Management Guidelines (AMGs) and revisions to Emergency Procedure Guidelines (EPGs). The following modifications are based on implementation of such generic activity.

A.4.1.1 Severe Accident EPGs/AMGs

The symptom based EPGs, were developed by the BWR Owners Group following the accident at Three Mile Island, Unit 2. Currently the EPGs are under revision and accident management guidelines (AMGs) are being developed for severe accidents. These should provide a significant improvement which reduces the likelihood of a severe accident. Elements of these guidelines (such as containment pressure and temperature control guidelines) also deal with mitigating the effects of accidents.

In the ABWR PRA, Emergency Operating Procedures (EOPs) are based on these guidelines. Additional extensions of the EPGs and EOPs could be made to address arrest of a core melt, emergency planning, radiological release assessment and other areas related to severe accidents.

Since the existing EPGs cover preventive actions and some mitigative actions, the incremental benefit of this item would be primarily mitigative. It was judged that the reliability of manual actions associated with mitigation could be improved by 10%, especially in use of core melt arrest processes. Failure rates for manually initiated mitigative systems were decreased by 10%, to estimate the benefit. The resulting offsite risk reduction is about 0.015 person-rem over 60 years.

A.4.1.2 Computer Aided Instrumentation

Computer aided artificial intelligence can be added which provides attention to risk issues in man-machine interfaces. Significant computer assisted display and plant status monitoring is already part of the ABWR control room design. Additional artificial intelligence could be designed which would display procedural options for the operator to evaluate during severe accidents. The system would be an extension of ERIS to provide human engineered displays of the important variables in the EPGs and AMGs.

Operator actions are made significantly more reliable by new features such as Emergency Procedure Guidelines, Safety Plant Parameter Displays (SPDS), and training on simulators. If the improvements described in Subsection A.4.1.1 are assumed to be implemented, the incremental benefit of additional improvements is expected to be low. The reliability of manually initiated preventive systems was increased by 10% to estimate the benefit. The estimated incremental benefit over severe accident EPGs (Subsection A.4.1.1) is about 3% in core damage frequency (CDF). Because the improvement affects all release cases, the incremental benefit is about 0.01 person-rem.

A.4.1.3 Improved Maintenance Procedures/Manuals

For the GE scope of supply this item would provide additional information on the components important to the risk of the plant. As a result of improved maintenance manuals and information it would be expected that increased reliability of the important equipment would occur. This item would be a preventative improvement which would address several system or components to different degrees.

Based on a 10% improvement in the reliability of the High Pressure Core Flooder (HPCF), Reactor Core Isolation Cooling (RCIC), Residual Heat Removal (RHR) and Low Pressure Core Flooder (LPFL) systems, the CDF is reduced by about 9% which has a corresponding estimated person-rem reduction of about 0.016.

A.4.2 Decay Heat Removal

Significant improvements in the reliability of ABWR high pressure systems have been made. Among these are RCIC restart (NUREG 0737, II.K.3.13) and isolation reliability improvements (NUREG 0737, II.K.3.15). Additionally, the redundant HPCF is an improvement over early product lines which used the single HPCF system.

A.4.2.1 Passive High Pressure System

This concept would provide additional high pressure capability to remove decay heat through a diverse isolation condenser type system. Such a system would have the advantage of removing not only decay heat, but containment heat if a similar system to that under consideration for the Simplified BWR (SBWR) is employed.

The benefit of this system would be equivalent to an additional diverse RCIC system in addition to an additional containment heat removal system. The added system was assumed to be 90% reliable, designed to operate independent of offsite power and to be capable of in-vessel core melt arrest. Based on a reduction in the RCIC failure rate, the benefit is estimated at about 0.069 person-rem averted.

A.4.2.2 Improved Depressurization

This item would provide an improved depressurization system which would allow more reliable access to low pressure systems. Additional depressurization capability may be achieved through manually controlled, seismically protected, air powered operators which permit depressurization to be manually accomplished in the event of loss of DC control power or control air events.

The ABWR high pressure core damage events represent about 28% of the total core damage frequency, but about 46% of the offsite exposure risk. The success of manual initiation was assumed to be improved by 50% and therefore the depressurization failure rate was reduced by a factor of 2. Based on this estimate of benefit offsite person-rem is reduced by about 23% and the estimated benefit is about 0.042 person-rem.

A.4.2.3 Suppression Pool Jockey Pump

This modification would provide a small makeup pump to provide low pressure decay heat removal from the Reactor Pressure Vessel (RPV) using suppression pool water as a source. The return path to the suppression pool would be through existing piping such as shutdown cooling return lines.

The benefit of this modification would be similar to that provided by the firewater injection and spray capability, but it would have the advantage that long term containment inventory concerns would not occur.

If the system could make low pressure coolant makeup systems 10% more reliable, significant reductions in CDF would not be achieved because other low pressure systems are already highly reliable. The estimated benefit is that CDF is reduced 2% and the averted risk would be 0.002 person-rem.

A.4.2.4 Safety-Related Condensate Storage Tank

The current ABWR design consists of a standard non-seismically qualified Condensate Storage Tank (CST). This modification would upgrade the structure of the CST such that it would be available to provide makeup to the reactor following a seismic event.

This modification only benefits the risks of core damage following seismic events. However, because the suppression pool provides an alternate suction source and the HCLPF for the suppression pool is relatively high (Appendix 19I of the ABWR SSAR), the dominant failure modes are not limited by water availability. Therefore the benefit of this modification is considered small. A benefit of 0.01 person-rem averted was arbitrarily chosen for an upgraded CST.

A.4.3 Containment Capability

The ABWR containment is designed for about 45 psig internal pressure and includes a containment rupture disk which would relieve excessive pressure if it develops during a severe accident. By providing the release point from the wetwell airspace, mitigation of releases are achieved through scrubbing of the fission products in the suppression pool.

A.4.3.1 Larger Volume Containment

This modification would provide a larger volume containment as a means to mitigate the effects of severe accidents. By increasing the size the containment could be able to absorb additional noncondensible gas generation and delay activation of the containment rupture disk or early containment failure.

This item would mitigate the consequence of an accident by delaying the time before the severe accident source term is released and allowing more time for radioactive decay and recovery of systems. However, if recovery does not occur, eventual release is not prevented and if operation of the containment overpressure rupture disk does not occur, ultimately the containment will fail due to the long term pressurization caused by core concrete interaction and steam generation.

If sequences involving drywell head failure were eliminated (Cases 3, 6, 7, 8 and 9), the offsite risks would be reduced by about 82% and about 0.15 person-rem would be averted.

A.4.3.2 Increased Containment Pressure Capacity

The design pressure of the ABWR containment is 45 psig. The containment rupture disk pressure and ultimate capability are significantly higher. By increasing the ultimate pressure capability of the containment (including seals), the effects of a severe accident could be reduced or eliminated by delaying the time of release. If the strength exceeded the maximum pressure obtainable in a severe accident, only normal containment leakage would result.

This modification would mitigate the event, not change the core damage frequency and the increased pressure capability may not be sufficient to contain the long term pressurization caused by core concrete interaction and steam generation. However, if it were able to prevent all severe source term release except for normal containment leakage, the person-rem risk would be about 0.02 person-rem/60 years. Therefore, the benefit would be about 0.16 person-rem.

A.4.3.3 Improved Vacuum Breakers

The ABWR design contains single vacuum breaker valves in each of eight drywell to wetwell vacuum breaker lines. The PRA included failure of vacuum breakers in Case 2 assuming operation of wetwell spray. This modification would reduce the probability of a stuck open vacuum breaker by making the valves redundant in each line and eliminate the need for operator action.

If Case 2 sequences were eliminated, the benefit of this modification would be about 0.00004 person-rem averted.

A.4.3.4 Improved Bottom Head Penetration Design

The ABWR design includes a 2-inch stainless steel drainline from the bottom of the RPV which is used to prevent thermal stratification in the RPV during operation and to provide cleanup of the bottom head by the CUW system. A carbon steel transition piece connects the drain line to the RPV. During a severe accident this transition piece may be susceptible to melting and may provide the earliest path for release of molten core material from the RPV to the containment.

The penetrations for the fine motion control rod drives in the ABWR also may provide a pathway for release from the RPV following a severe accident. Failure of the internal blowout supports on the lower core plate, provided to eliminate the support structure in current generation BWRs, and welds of the drives at the bottom of the vessel may allow the CRDs to be partially ejected into the drywell during the severe accident which would provide a small pathway for release to the containment.

The modification is to change the transition piece material to Inconel or Stainless Steel which has a higher melting point. By so doing, additional time would be available for recovery of core cooling systems. This modification also would establish external welds or restraints on the CRDs external to the vessel so that the drives would not be ejected following failure of the internal welds. The concept would be to make such external welds and supports small enough that the benefit is not lost from eliminating the support beams in current generation BWRs. The benefit of these modifications would be to reduce the probability of in-vessel arrest failure (NO IV). Based on consideration of the heatup rate of the bottom head, it has been estimated that making these changes could provide up to two hours additional time for recovery of systems. It is estimated, based on engineering judgment, that this time could result in the in-vessel arrest failure probabilities being reduced by a factor of two. The resulting benefit is about 0.057 person-rem averted.

A potential negative aspect of the modifications is that RPV failure could occur at another unknown location such as the bottom head itself. Although the time of vessel failure would be extended, the failure mode from these other locations could be potentially more energetic and lead to unevaluated consequences.

A.4.4 Containment Heat Removal

The ABWR design contains 3 divisions of suppression pool cooling and provisions for a containment rupture disk for decay heat removal. In addition, modifications have been made to use the CUW heat exchangers to the maximum extent possible. Consequently, loss of containment heat removal events contribute only 0.1% of the total core damage frequency and offsite exposures. Additional modifications are not likely to show substantial safety benefits.

A.4.4.1 Larger Volume Suppression Pool

This item would increase the size of the suppression pool so that the heatup rate in the pool is reduced. The increased size would allow more time for recovery of a heat removal system.

Since this modification primarily affects LHRC events (Table A-2), the maximum benefit would be elimination of the LHRC contribution to the Case 9 sequences. These events are mitigated by the containment rupture disk and only contribute about 0.0002 person-rem to the base case risk. The assessed maximum benefit is therefore about 0.0002 person-rem.

A.4.5 Containment Atmosphere Mass Removal

The ABWR design contains a containment rupture disk which provides containment overpressure protection from the wetwell airspace and utilizes the suppression pool scrubbing feature of the suppression pool to reduce the amount of radioactive material released. One additional modification was considered.

A.4.5.1 Low Flow Filtered Vent

Some BWR facilities, especially in Europe, recently have added a filter system external to the containment to further reduce the magnitude of radioactive release. The systems typically use a multi-venturi scrubbing system to circulate the exhaust gas and remove particulate material. In the ABWR because of the suppression pool scrubbing capability, a significant safety improvement is not expected due to this modification.

The release of radioactive isotopes from the ABWR following severe accidents occurs through the containment rupture disk for Cases 1, 2 and 5. These sequences total about 8% of the exposure risk. The remaining sequences involve drywell head failure or early containment failure which would not be affected by this modification. The maximum benefit of the external vent system is therefore about 0.014 person-rem assuming perfect initiation of the filtered containment vent system.

A.4.6 Combustible Gas Control

No additional modifications to the ABWR were identified in this group.

A.4.7 Containment Spray Systems

A.4.7.1 Drywell Head Flooding

This concept would provide intentional flooding of the upper drywell head such that if high drywell temperatures occurred, the drywell head seal would not fail. Additionally, if the seal were to fail due to overpressurization of the drywell, some scrubbing of the released fission products would occur. This system would be designed to operate passively or use an AC-independent water source.

If an extension of the fire pump to drywell spray crosstie were considered for manual initiation of upper head flooding, additional reduction in the high temperature containment failure sequences (Case 8) would result. Additionally, a reduction in the high consequence drywell head failure sequences (Cases 6 and 7) could be achieved. If Case 8 sequences were eliminated and Case 6 and 7 source terms were reduced to a level similar to Case 3, the conservative benefit would be 0.12 person-rem. The estimated benefit of this is about 0.06 person-rem assuming a 50% reliability of initiation.

A.4.8 Prevention Concepts

The ABWR design contains an additional division of high pressure makeup capability to improve its capability to prevent severe accidents other features such as the fire pump injection capability and the combustion gas turbine have been included in the design to enhance the plant capability to prevent core damage. The following additional concepts were considered:

A.4.8.1 Additional Service Water Pumps

This item addresses a reduction in the common cause dependencies through such items as improved manufacturer diversity, separation of equipment and support systems such as service water, air supplies, or heating and ventilation (HVAC). The HPCF, RCIC, and LPFL pumps are diverse in the ABWR design since they are either supplied by different manufacturers or have different flow characteristics. Equipment is separated in the ABWR design in accordance with Regulatory Guide 1.75. Thus, no further improvement is expected with regard to separation.

A reduction in common cause dependencies from support systems such as service water systems, could conceivably reduce the plant risk through an improvement in system reliability. The concept for this item would be to provide an additional cooling water system capable of supporting each of the four divisional systems identified above.

The current design provides support to these systems from one of three divisions. Thus, the effect of this change would be to include a diverse and additional support system. In addition, diversity in instrumentation which controls these systems could be included so that redundant indication and trip channels would rely on diverse instrumentation.

A 10% increase in the reliability of the four systems was assumed which is the same improvement that may be derived from improved maintenance (Subsection A.4.1.3). This results in an estimated benefit of about 0.016 person-rem.

A.4.9 AC Power Supplies

The current ABWR electrical design is improved through application of a gas-turbine generator to augment the offsite electrical grid. The following concepts were considered for additional onsite power supplies.

A.4.9.1 Steam Driven Turbine Generator

A steam driven turbine generator could be installed which uses reactor steam and exhausts to the suppression pool. The system would be conceptually similar to the RCIC system with the generator connected to the offsite power grid.

The benefit of this item would be similar to the addition of another gas turbine generator, but would be somewhat less due to the relative unreliability of the steam turbine compare·' with a diesel generator and its unavailability after the RPV is depressurized. If it were sized l ge enough, it could have the advantage of providing power to additional equipment.

If the system has a 80% availability for all events, the benefit is similar to an 80% reduction in the diesel generator common mode failure rate. Evaluation of the PRA indicates that the resulting benefit is about 0.052 person-rem.

A.4.9.2 Alternate Pump Power Source

The ABWR provides separate diesel driven power supplies to the HPCF and LPFL pumps. Offsite power supplies the feedwater pumps. This modification would provide a small dedicated power source such as a dedicated diesel or gas turbine for the feedwater, or condensate pumps so that they do not rely on offsite power.

The benefit would be less dependence on low pressure systems during loss of offsite power events and station blackout events. If the feedwater system were made to be 90% available during loss of offsite power events and station blackouts, the benefit would be similar to adding an additional RCIC system (Subsection A.4.2.1). The resulting benefit would be about 0.069 person-rem.

A.4.10 DC Power Supplies

The ABWR contains 4 DC divisions with sufficient capacity to sustain 8 hours of station blackout (with some load shedding). This represents an improvement over current operating plant designs.

A.4.10.1 Dedicated DC Power Supply

This item addresses the use of a diverse DC power system such as an additional battery or fuel cell for the purpose of providing motive power to certain components. Conceptually a fuel cell or separate battery could be used to power a DC motor/pump combination and provide high pressure RPV injection and containment cooling. With proper starting controls such a system could be sized to provide several days capability.

Providing a separate DC powered high pressure injection capability has a benefit of further reducing the station blackout and loss of offsite power event risks which represent about 75% of the total CDF, but only a small fraction of the offsite risk. If the effective unavailability of the RCIC is reduced by a factor of 10 due to the availability of a diverse system, one benefit would be similar to adding a power supply for feedwater (Subsection A.4.9.2) and the benefit would be about 0.069 person-rem.

A.4.11 ATWS Capability

The current ABWR design provides improvements in containment heat removal and detection of ATWS events to limit the impact of this class of events. The PRA indicates that ATWS events contribute about 0.1% of the core damage frequency (Table A-2) and about 17% of the offsite risk ((Case 9).

A.4.11.1 ATWS Sized Vent

This modification would be available to remove reactor heat from ATWS events in addition to severe accidents and Class II events. It would be similar to the containment rupture disk (which is currently sized to pass reactor power consistent with that generated during RCIC injection), but it would be of the larger size required to pass the additional steam associated with LPFL injection. The system would need to be manually initiated.

The benefit of this venting concept is to prevent core damage and to reduce the source term available for release following ATWS events. The evaluation shows that an ATWS sized vent manually initiated with a 100% reliability would have a maximum benefit of reducing the offsite dose by about 0.03 person-rem by reassigning the consequences from Case 9 to Case 1.

A.4.12 Seismic Capability

The current ABWR is designed for a Safe Shutdown Earthquake of 0.3g acceleration. The seismic margins analysis (Appendix 19I of the ABWR SSAR) addresses the margins associated with the seismic design and concludes that there is a 95% confidence that existing equipment has less than a 5% probability of failure at twice the SSE level. This capability is considered adequate for the ABWR design and no additional changes are considered.

A.4.13 System Simplification

This item is intended to address system simplification by the elimination of unnecessary interlocks, automatic initiation of manual actions or redundancy as a means to reduce overall plant risk. Elimination of seismic and pipe whip restraints is included in the concept.

While there are several examples of redundant systems, valves and features on the ABWR design which could conceivably be simplified, there are several areas in which the ABWR design already has been improved and simplified, especially in the area of controls and logic. System interactions during accidents were included in this category. One area was identified in which simple modification of an existing system could provide some benefit.

A.4.13.1 Reactor Building Sprays

This concept would use the firewater sprays in the reactor building to mitigate releases of fission products into the reactor building following an accident. The concept would require additional valves and nozzles, separate from the fire protection fusible links, to spray in areas vulnerable to release, such as near the containment overpressure relief line routing.

The benefit of this modification could be to reduce the impact of events which do not involve the operation of the containment rupture disk. Such events release fission products from the containment into the reactor building. Releases from normal containment leakage and cases 3, 6, 7, 8 and case 9 sequences could potentially be reduced. If 10% of these releases from these cases were arbitrarily mitigated by this method, the benefit would be about 1.7E-04 person-rem.

A.4.14 Core Retention Devices

Core retention features are incorporated into the ABWR Design. As discussed in Subsection 19E.2.2(paragraph FS) of the ABWR SSAR, if a severe accident has resulted in a loss of RPV integrity, accident management guidance specifies that drywell sprays be initiated which will cause the suppression pool to overflow into the lower drywell after a few hours and quench the debris bed. After the molten core has been quenched, no further ablation of concrete is expected and the decay heat can be removed by normal containment cooling methods such as suppression pool cooling. If sprays can not be initiated, the Lower Drywell Flooder System described in Subsection 9.5.12 of the ABWR SSAR cools a debris bed by flooding over the molten core in the lower drywell with water from the suppression pool. This system is similar to the Post Accident Flooding concept included in Reference A-4. One additional concept from Reference A-4 is included.

A.4.14.1 Flooded Rubble Bed

This concept consists of a bed of refractory pebbles which fill the lower drywell cavity and are flooded with water. The bed impedes the flow of molten corium and increases the available heat transfer area which enhances debris coolability. The use of thoria ($ThO2$) pellets in a multiple layer geometry has been shown to stop melt penetration; thus, preventing core-concrete interaction. Drawbacks to using thorium dioxide include cost, toxicity, and the radiological impact of radon gas release into the lower drywell via the radioactive decay of thorium. Other refractories such as alumina slow corium penetration but may fail to stop core-concrete contact. Other refractories may be susceptible to chemical attack by the corium and may melt at lower temperatures. Pebbles composed of refractories other than thoria also may be susceptible to floating because they have lower density than the corium. A major drawback common to all flooded rubble bed core retention systems is the need for further experimental testing in order to validate the concept in BWR applications.

The benefit of this modification lies in the potential elimination of core-concrete interaction and a corresponding decrease in non-condensable gas generation. Attachment 19EC to Appendix 19E of the ABWR SSAR indicates a 90% certainty that debris on a concrete floor covered with water will be coolable in the current ABWR design.

Only sequences in which no liquid injection to the drywell occurs will result in core-concrete interaction. A conservative estimate of the benefit of this concept over the existing design would be elimination of sequences with core-concrete interaction except those with containment cooling failure. A review of Subsection 19E.2 of the ABWR SSAR indicates that this would effect about 1% of Cases 1, 6 and 7. This corresponds to about 0.001 person-rem averted.

A.5 COST IMPACTS OF POTENTIAL MODIFICATIONS

As discussed in Subsection A.1.3.1, rough order of magnitude costs were assigned to each modification based on the costs of systems determined by GE. These costs represent the incremental costs that would be incurred in a new plant rather than costs that would apply on a backfit basis. Credit for the onsite costs averted by the modification are discussed in Subsection A.1.3.2. For each modification which reduces the core damage frequency an estimate of the impact was made and then applied to the potential averted offsite cost. This section summarizes the cost basis for each of the modification evaluated in Section A.4. This basis is generally the cost estimate less the credit for onsite averted costs. Table A-6 summarizes the results.

The costs were biased on the low side, but all known or reasonably expected costs were accounted for in order that a reasonable assessment of the minimum cost would be obtained. Actual plant costs are expected to be higher than indicated in this evaluation. All costs are referenced to 1991 U.S. dollars based on changes in the Consumer Price Index.

A.5.1 Accident Management

A.5.1.1 Severe Accident EPGs/AMGs

The cost of extending the EPGs would be largely a one-time cost which should be prorated over several plants if accomplished by the BWROG. Current industry activity is addressing this as part of Accident Management Guidelines (AMG). If plant specific, symptom based, severe accident emergency procedures were to be prepared based on AMGs, the cost would be at least $600,000 for plant specific modifications to EOPs.

A.5.1.2 Computer-Aided Instrumentation

Additional software and development costs associated with modifying existing Safety Plant Display Systems are estimated to cost at least $600,000 for a new plant. This estimate is based on assumed additions of isolation devices to transmit data to the computer and in-plant wiring. Because this modification reduces the frequency of core damage events, a present worth of $400 onsite costs are averted and the cost basis is $599,600.

A.5.1.3 Improved Maintenance Procedures/Manuals

The cost of at least $300,000 would be required to identify components which should receive enhanced maintenance attention and to prepare the additional detailed procedures or recommended information beyond that currently planned. Credit for reduction in onsite costs reduces the cost basis to $299,000.

A.5.2 Decay Heat Removal

A.5.2.1 Passive High Pressure System

The cost of an additional high pressure system for core cooling would be extensive since it would not only require additional system hardware which would cost at least $1,200,000, but it would also require additional building costs for space available for the system. Assuming the system could be located in the reactor building without increasing its height, building costs are estimated to be another $550,000. The credit for averted onsite costs is about $6,000 which brings the cost basis to $1,744,000.

A.5.2.2 Improved Depressurization

The cost of the additional logic changes, pneumatic supplies, piping and qualification was estimated for the GESSAR II design (Reference A-1). A similar cost would be expected for the ABWR design. The cost is estimated to be at least $600,000 for an improved system for depressurization. This estimate assumes no building space increase for the added equipment. The credit for averted onsite costs was evaluated to be $1,400 which makes the cost basis $598,600.

A.5.2.3 Suppression Pool Jockey Pump

The cost of an additional small pump and associated piping is estimated at more than $60,000 including installation of the equipment. It is assumed that increases in power supply capacity and building space are not required. Controls and associated wiring could cost an additional $60,000 for a total cost of at least $120,000. A credit of $200 for averted onsite costs makes the cost basis $119,800.

A.5.2.4 Safety Related Condensate Storage Tank

Estimating the cost of upgrading the CST structure to withstand seismic events requires a detailed structural analysis and resultant material. It is judged that the final cost increase would be in excess of $1,000,000. No credit for onsite cost averted was assumed for this modification.

A.5.3 Containment Capability

A.5.3.1 Larger Volume Containment

Doubling the containment volume requires an increase in the concrete and rebar. If structural costs of the containment can be made for \$1,200/ft^3, doubling the containment volume without increasing its height, the cost would be at least \$8,000,000. This estimate does not include reanalysis and other documentation costs. Since this modification is mitigative, no credit for onsite averted costs was assumed.

A.5.3.2 Increased Containment Pressure Capacity

The cost of a stronger containment design would be similar in magnitude to increasing its size (Subsection A.5.3.1). If the costs are primarily due to denser rebar required during installation and additional analysis, an estimate of at least \$12,000,000 could be required. Since this modification is mitigative, no credit for onsite averted costs was assumed.

A.5.3.3 Improved Vacuum Breakers

The cost of redundant vacuum breakers including installation and hardware is estimated at more than \$10,000 per line. Instrumentation associated with this modification is not included. For the eight lines the cost of this modification is more than \$100,000. Since this modification is mitigative, no credit for onsite averted costs was assumed.

A.5.3.4 Improved Bottom Penetration Design

The cost increase of using a stainless or inconel transition piece as opposed to carbon steel would be expected to be small in comparison to the engineering and documentation change costs associated with the change. Costs, associated with external welds and support for the CRDs is estimated to be at least \$1000 per drive. In addition, about \$500,000 of analysis would be required to develop the changes. This would dominate the cost of this modification when applied to all 205 drives. Such changes are estimated to be at least \$750,000.

Since this modification is mitigative, no credit for averted onsite costs applies.

A.5.4 Containment Heat Removal

A.5.4.1 Larger Volume Suppression Pool

This concept would result in similar costs as item Subsection A.5.3.1 for providing a larger containment. An estimate of \$8,000,000 is assigned to this item.

A.5.5 Containment Atmosphere Mass Removal

A.5.5.1 Low Flow Filtered Vent

The cost of added equipment associated with the FILTRA system (excluding a test program) was estimated to be about $5,000,000 in Reference A-4. Although a detailed estimate was not prepared for the ABWR, an estimate of $3,000,000 has been assumed for the purpose of this evaluation.

Since this modification is mitigative, no credit for averted onsite costs applies.

A.5.6 Combustible Gas Control

No additional modifications to the ABWR were identified in this group.

A.5.7 Containment Spray Systems

A.5.7.1 Drywell Head Flooding

An additional line to flood the drywell head using existing firewater piping would be a relatively inexpensive addition to the current system. Instrumentation and controls to permit manual control from the control room would be needed. It is estimated that the total modification cost would be at least $100,000 for the engineering, piping, valves and cabling.

Because this modification is mitigative, no credit for averted onsite costs has been applied.

A.5.8 Prevention Concepts

A.5.8.1 Additional Service Water Pump

The use of diverse instrumentation would not presumably have a significant equipment cost, but there would be an increased cost of maintenance and spare parts due to less interchangeability and less standardization of procedures.

These costs, however, are probably low in comparison with the extra support systems for air supply and service water. Equipment, power supplies and structural changes to include these new systems are estimated to cost at least $6,000,000. A small credit for averted onsite costs makes the cost basis for this item $5,999,000, based on the benefits discussed in Subsections A.4.1.3 and A.5.1.3.

A.5.9 AC Power Supplies

A.5.9.1 Steam-Driven Turbine Generator

The cost of the system should be similar to that for the RCIC system, but additional cost would be needed for structural changes to the reactor building plus the generator and its controls. This item is expected to cost at least $6,000,000.

With credit for averted onsite costs, the cost basis for this item becomes $5,994,300.

A.5.9.2 Alternate Pump Power Source

A typical feedwater pump for an ABWR sized plant could require a 4000 kWe sized generator, at $300 per kWe, a separate diesel generator and the supporting auxiliaries could cost at least $1,200,000. This cost would include wiring and installation of the alternate generator, but does not assume additional structural costs.

With credit for averted onsite costs, the cost basis for this item becomes $1,194,000.

A.5.10 DC Power Supplies

A.5.10.1 Dedicated DC Power Supply

Fuel cells are largely a developmental technology, at least in the large size range required for this application. In addition the process involves some risk of fire. To address these concerns a cost of at least $6,000,000 would be expected. A separate battery would be less expensive than fuel cells, but would involve additional space requirements which could make this modification more expensive than adding a diesel generator as discussed in Subsection A.5.9.2.

A battery bank capable of supplying 400 kWe would be about 50 times larger in capacity than the emergency batteries. This number of batteries would require at least 5,000 ft^2 of space, assuming extensive stacking and without concern for seismic response. At $500/ft^2 construction cost, the additional space required would amount to $2,500,000 for this modification. Additional costs would be required for DC pumps, cabling and instrumentation and controllers. A total cost would be at least $3,000,000.

A.5.11 ATWS Capability

A.5.11.1 ATWS Sized Vent

Larger piping and additional training would be required to extend the existing rupture disk feature to be available during an ATWS event. Additional instrumentation and cabling would be required to make the vent operable from the control room. It is estimated that the incremental cost would be at least $300,000.

A.5.12 Seismic Capability

No modifications were considered for this group.

A.5.13 System Simplification

A.5.13.1 Reactor Building Sprays

The cost of this modification is judged to be similar to the concept of drywell head flooding (Subsection A.5.5.1) if it only involves piping and valves which are tied into the firewater system. An estimate of $100,000 has been assigned to this item.

Onsite cleanup costs also could be affected by this modification. If the cleanup costs were eliminated an averted cost would conservatively be about $5,000.

A.5.14 Core Retention Devices

A.5.14.1 Flooded Rubble Bed

Reference A-4 estimated that the refractory material needed for this modification would cost approximately $1,000/lb. If the lower drywell were filled with about 1.5 ft of this material, which would remain well below the service platform, at least 1250 ft^3 of material would be required. If it weighs 15 lb/ft^3, the material cost alone would amount to $18,750,000.

A.6 EVALUATION OF POTENTIAL MODIFICATIONS

A ranking of the modifications by $/person-rem averted is shown in Table A-7 based on the results and estimates provided in Sections A.4 and A.5.

The lowest cost/person-rem averted modification is more than 1600 times the target criteria of $1,000 per person-rem averted. Clearly none of the modifications is justifiable on the basis of costs for person-rem averted. This can be attributed to the low probability of core damage in the ABWR with the modifications to reduce risk already installed.

A.7 SUMMARY OF CONCLUSIONS

Potentially attractive modifications were identified from previous evaluations of potential prevention and mitigation concepts applicable during severe accidents and discussion with the NRC staff. Potential modifications were reviewed to select those which are applicable to the ABWR design and which have not already been implemented in the design. Of these modifications, twenty one were selected for additional review.

The low level of risk in the ABWR is demonstrated by the total 60 year offsite exposure risk of 0.269 person-rem. At this level only modifications which cost less than $269 can be justified.

Based on this low level no modifications are justified for the ABWR. Based on the PRA results, none of the modifications provided a substantial improvement in plant safety.

A.8 REFERENCES

A-1 Evaluation of Proposed Modifications to the GESSAR II Design, NEDE 30640 (Proprietary), June 1984.

A-2 Supplement to the Final Environmental Statement - Limerick Generating Station, Units 1 and 2, NUREG-0974 Supplement, August 16, 1989

A-3 Issuance of Supplement to the Final Environmental Statement- Comanche Peak Steam Electric Station, Units 1 and 2, NUREG 0775 Supplement, December 15, 1989

A-4 Survey of the State of the Art in Mitigation Systems, NUREG/CR-3908, R&D Associates, December 1985

A-5 Assessment of Severe Accident Prevention and Mitigation Features, NUREG/CR-4920, Brookhaven National Laboratory, July 1988.

A-6 Design and Feasibility of Accident Mitigation Systems for Light Water Reactors, NUREG/CR-4025, R&D Associates, August 1985

A-7 Severe Accident Risks: An Assessment for Five US Nuclear Power Plants, NUREG 1150, January 1991.

A-8 Technical Guidance for Siting Criteria Development, NUREG/CR-2239, Sandia National Laboratories, December 1982.

Table A-1
Radiological Consequences of ABWR Accident Sequences

Case	Probability (Event/year)*	Whole Body Exposure, 50 mile (person-rem)	Cumulative Exposure Risk (per-rem/60 yr)
NCL	1.3E-07	9.60E3	0.075
1	2.1E-08	1.38E4	0.017
2	7.8E-11	8.33E3	0.00004
3	0	3.71E5	0.000
4	0	2.06E5	0.000
5	7.5E-12	9.34E4	0.00004
6	3.1E-12	2.42E6	0.0004
7	3.9E-10	2.73E6	0.064
8	4.1E-10	3.20E6	0.079
9	1.7E-10	3.31E6	0.034
		Total:	0.269

* Sequences with probabilities of occurrence less than 1E-9 per year are considered remote and speculative.

Table A-2
Core Damage Frequency Contributors*

Event Sequence

Init. Event	1A	1B1	1B2	1B3	1D	II	IIID	IV	Total	% Cont.
Scram	1.1E-08				4.3E-10	9.5E-13			1.1E-08	7.3
Turbine Trip	6.8E-09				2.7E-10	3.7E-11			7.1E-09	4.5
Isolation	1.8E-08				7.1E-10	1.1E-11			1.9E-08	11.9
LOOP2	4.1E-09				1.5E-11	4.2E-13			4.1E-09	2.6
LOOP8	2.4E-09				9.6E-12	1.4E-12			2.4E-09	1.5
LOOP8+	5.8E-10				1.1E-09	6.0E-11			1.7E-09	1.1
SBO2	6.6E-12				6.7E-08				6.7E-08	42.9
SBO8		2.6E-08							2.6E-08	16.7
SBO8+			1.5E-08	8.9E-10					1.6E-08	10.3
IORV	1.1E-09				2.0E-10	9.5E-13			1.3E-09	0.8
SB LOCA							2.5E-10		2.5E-10	0.2
ATWS								1.5E-10	1.5E-10	0.1
TOTAL	4.4E-08	2.6E-08	1.5E-08	8.9E-10	7.0E-08	1.1E-10	2.5E-10	1.5E-10	1.57E-07	100

Offsite Release Group

	LCHP	SBRC	LCLP	LHRC	LBLC	ATWS	Total Case
Case 1	3.4E-09	7.9E-10	1.6E-08		5.1E-11		2.0E-08
Case 2			7.8E-11				7.8E-11
Case 3	1.3E-12						1.3E-12
Case 4							0
Case 5					6.3E-12		6.3E-12
Case 6	1.2E-10						1.2E-10
Case 7	1.1E-10		2.6E-10				3.70E-10
Case 8	2.1E-10						2.1E-10
Case 9				1.1E-12		1.5E-10	1.5E-10
NCL (N)	4.0E-08	1.5E-08	8.0E-08		2.0E-10		1.4E-07
Total	4.4E-08	1.6E-08	9.6E-08	1.1E-12	2.5E-10	1.5E-10	1.57E-07
Contrib. %	28.1	10.3	61.4	0.122	0.2	0.1	100

* SAMDAs include both preventive and mitigative design alternatives

Table A-3
Modifications Considered

Modification	Category
1. ACCIDENT MANAGEMENT	
a. Severe Accident EPGs/AMGs	2
b. Computer Aided Instrumentation	2
c. Improved Maintenance Procedures/Manuals	2
d. Preventive Maintenance Features	4
e. Improved Accident Management Instrumentation	4
f. Remote Shutdown Station	1
g. Security System	1
h. Simulator Training for Severe Accident	4
2. REACTOR DECAY HEAT REMOVAL	
a. Passive High Pressure System	2
b. Improved Depressurization	2
c. Suppression Pool Jockey Pump	2
d. Improved High Pressure Systems	1
e. Additional Active High Pressure System	1
f. Improved Low Pressure System (Firepump)	1
g. Dedicated Suppression Pool Cooling	1
h. Safety Related Condensate Storage Tank	2
i. 16 hour Station Blackout Injection	4
j. Improved Recirculation Model	4
3. CONTAINMENT CAPABILITY	
a. Larger Volume Containment	2
b. Increased Containment Pressure Capacity	2
c. Improved Vacuum Breakers	2
d. Increased Temperature Margin for Seals	1
e. Improved Leak Detection	1
f. Suppression Pool Scrubbing	1
g. Improved Bottom Penetration Design	2

Table A-3 (Continued)

Modification	Category
4. CONTAINMENT HEAT REMOVAL	
a. Larger Volume Suppression Pool	2
b. CUW Decay Heat Removal	1
c. High Flow Suppression Pool Cooling	1
d. Passive Overpressure Relief	1
5. CONTAINMENT ATMOSPHERE MASS REMOVAL	
a. High Flow Unfiltered Vent	3
b. High Flow Filtered Vent	3
c. Low Flow Vent (Filtered)	2
d. Low Flow Vent (Unfiltered)	1
6. COMBUSTIBLE GAS CONTROL	
a. Post-Accident Inerting System	3
b. Hydrogen Control by Venting	3
c. Pre-inerting	1
d. Ignition Systems	3
e. Fire Suppression System Inerting	3
7. CONTAINMENT SPRAY SYSTEMS	
a. Drywell Head Flooding	2
b. Containment Spray Augmentation	1
8. PREVENTION CONCEPTS	
a. Additional Service Water Pump	2
b. Improved Operating Response	1
c. Diverse Injection System	4
d. Operating Experience Feedback	1
e. Improved MSIV/SRV Design	1
9. AC POWER SUPPLIES	
a. Steam Driven Turbine Generator	2
b. Alternate Pump Power Source	2
c. Deleted	
d. Additional Diesel Generator	1

Table A-3 (Continued)

	Modification	Category
9.	(Continued)	
	e. Increased Electrical Divisions	1
	f. Improved Uninterruptable Power Supplies	1
	g. AC Bus Cross-ties	1
	h. Gas Turbine	1
	i. Dedicated RHR (bunkered) Power Supply	4
10.	DC POWER SUPPLIES	
	a. Dedicated DC Power Supply	2
	b. Additional Batteries/Divisions	4
	c. Fuel Cells	4
	d. DC Cross-ties	1
	e. Extended Station Blackout Provisions	1
11.	ATWS CAPABILITY	
	a. ATWS Sized Vent	2
	b. Improved ATWS Capability	1
12.	SEISMIC CAPABILITY	
	a. Increased Seismic Margins	1
	b. Integral Basemat	3
13.	SYSTEM SIMPLIFICATION	
	a. Reactor Building Sprays	2
	b. System Simplification	1
	c. Reduction in Reactor Bldg Flooding	1
14.	CORE RETENTION DEVICES	
	a. Flooded Rubble Bed	2
	b. Reactor Cavity Flooder	1
	c. Basaltic Cements	1

Table A-4
Modifications Evaluated

1. Accident Management	1a. Severe Accident EPGs/AMGs 1b. Computer Aided Instrumentation 1c. Improved Maintenance Procedures/Manuals
2. Decay Heat Removal	2a. Passive High Pressure System 2b. Improved Depressurization 2c. Suppression Pool Jockey Pump 2d. Safety Related Condensate Storage Tank
3. Containment Capability	3a. Larger Volume Containment 3b. Increased Containment Pressure Capability 3c. Improved Vacuum Breakers 3d. Improved Bottom Head Penetration Design
4. Containment Heat Removal	4a. Larger Volume Suppression Pool
5. Containment Atmosphere Gas Removal	5a. Low Flow Filtered Vent
7. Containment Spray	7a. Drywell Head Flooding
8. Prevention Concepts	8a. Additional Service Water Pump
9. AC Power Supplies	9a. Steam Driven Turbine Generator 9b. Alternate Pump Power Source
10. DC Power Supplies	10a. Dedicated DC Power Supply
11. ATWS Capability	11a. ATWS Sized Vent
13. System Simplification	13a. Reactor Building Sprays
14. Core Retention Devices	14a. Flooded Rubble Bed

Table A-5
Summary of Benefits

Potential Improvement	Averted Risk Person-rem
1a. Severe Accident EPGs/AMGs	1.5E-2
1b. Computer Aided Instrumentation	1.0E-2
1c. Improved Maintenance Procedures/Manuals	1.6E-2
2a. Passive High Pressure System	6.9E-2
2b. Improved Depressurization	4.2E-2
2c. Suppression Pool Jockey Pump	0.2E-2
2d. Safety Related Condensate Storage Tank	1.0E-2
3a. Larger Volume Containment	15E-2
3b. Increased Containment Pressure Capability	16E-2
3c. Improved Vacuum Breakers	0.004E-2
3d. Improved Bottom Head Penetration Design	5.7E-2
4a. Larger Volume Suppression Pool	0.02E-2
5a. Low Flow Filtered Vent	1.4E-2
7a. Drywell Head Flooding	6.0E-2
8a. Additional Service Water Pump	1.6E-2
9a. Steam Driven Turbine Generator	5.2E-2
9b. Alternate Pump Power Source for high pressure systems	6.9E-2
10a. Dedicated DC Power Supply	6.9E-2
11a. ATWS Sized Vent	3.0E-2
13a. Reactor Building Sprays	1.7E-2
14a. Flooded Rubble Bed	0.1E-2

Table A-6
Summary of Costs

Potential Improvement	Estimated Minimum Cost
1a. Severe Accident EPGs/AMGs	$ 600,000
1b. Computer Aided Instrumentation	$ 599,600
1c. Improved Maintenance Procedures/Manuals	$ 299,000
2a. Passive High Pressure System	$ 1,744,000
2b. Improved Depressurization	$ 598,600
2c. Suppression Pool Jockey Pump	$ 119,800
2d. Safety Related Condensate Storage Tank	$ 1,000,000
3a. Larger Volume Containment	$ 8,000,000
3b Increased Containment Pressure Capability	$ 12,000,000
3c. Improved Vacuum Breakers	$ 100,000
3d. Improved Bottom Head Penetration Design	$ 750,000
4a. Larger Volume Suppression Pool	$ 8,000,000
5a. Low Flow Filtered Vent	$ 3,000,000
7a. Drywell Head Flooding	$ 100,000
8a Additional Service Water Pump	$ 5,999,000
9a. Steam Driven Turbine Generator	$ 5,994,300
9b. Alternate Pump Power Source	$ 1,194,000
10a. Dedicated DC Power Supply	$ 3,000,000
11a. ATWS Sized Vent	$ 300,000
13a. Reactor Building Sprays	$ 100,000
14a. Flooded Rubble Bed	$ 18,750,000

Table A-7
Summary of Results

Modification	Cost ($)/Person-rem Averted
7a. Drywell Head Flooding	$1,667
13a. Reactor Building Sprays	$5,882
11a. ATWS Sized Vent	$10,000
3d. Improved Bottom Penetration Design	$13,158
2b. Improved Depressurization	$14,252
9b. Alternate Pump Power Source	$17,304
1c. Improved Maintenance Procedures/Manuals	$18,688
2a. Passive High Pressure System	$25,275
1a. Severe Accident EPGs	$40,000
10a. Dedicated DC Power Supply	$43,478
3a. Larger Volume Containment	$53,333
2c. Suppression Pool Jockey Pump	$59,990
1b. Computer Aided Instrumentation	$59,960
3b. Increased Containment Pressure Capacity	$75,000
2d. Safety Related Condensate Storage Tank	$100,000
9a. Steam Driven Turbine Generator	$115,275
5a. Low Flow Filtered Vent	$214,286
8a. Additional Service Water Pump	$374,938
3e. Improved Vacuum Breakers	$2,500,000
14a. Flooded Rubble Bed	$18,750,000
4a. Larger Volume Suppression Pool	$40,000,000

Enclosure 6

MFN 10-342

CD – Supplemental Environmental Report – Amendment to ABWR Standard Design Certification