Docket No. 52-001 April 4, 1994

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

Dear Mr. Larkins:

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

The advance copy of the SER on the ABWR design was provided to the Advisory Committee on Reactor Safeguards (ACRS) in December 1993. This SER identified 14 open items remaining from the staff's review of GE Nuclear Energy's (GE's) application for design certification. In the March 9, 1994, letter, the staff provided the ACRS with revisions to the advance copy of the SER that addressed 13 of the open items. The staff identified Item F17.1.3-1, "Inspection of QA Program," as open.

The purpose of this letter is to: (1) provide the ACRS with the SER page changes that document the basis for resolution of Open Item 17.1.3-1, "Inspection of QA Program," and (2) inform the ACRS that the staff has reconsidered its March 9, 1994, position on Item F6.2.1.9-1, "Suppression Pool Strainer."

Enclosure 1 provides SER page markups documenting the resolution of Open Item F17.1.3-1. Enclosure 2 contains a revised page 6-29 from the advance copy of the SER and the March 30, 1994, staff letter to GE that discusses the rationale for rescinding the March 9, 1994, staff position on sizing the suppression pool strainers. Item F6.2.1.9-1 is now open.

Sincerely,
(Original signed by)
Dennis M. Crutchfield, Associate Director
for Advanced Reactors and License Renewal
Office of Nuclear Reactor Regulation

Enclosures:
1. Revised pages for ABWR SER
2. Letter of 3/30/94, to GE

cc w/o enclosures:
See next page

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on open Item 17.1.3-1

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NAME	PShea	DTang:sg	JNWilson	MVirgilio	ERossi	RBorchardt	DCrutchfield
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with comments

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Enclosure 1

Open Item F17.1.3-1

Description of Item	Inspection of QA program
Advance SER Page No.	17-5
Insert No.	G

which included Hitachi and Toshiba documents requested by the staff and translated into English, the auditors concluded that the design QA programs implemented by GE, Hitachi, and Toshiba met the applicable requirements of Appendix B to 10 CFR Part 50 and were acceptable for designing the ABWR. The inspection of the ABWR design process was performed in September 1993. The staff is currently evaluating GE's response and, therefore, this is an open item.

INSERT G

An applicant for a COL, when completing its detailed design and equipment selection during the COL design phase, will submit its QA program for the design phase for staff review. This will be in addition to the staff review of the COL applicant's QA program for both the construction and the operation of the facility. When the COL applicant's QA programs are submitted, whether they are the GE QA programs augmented with information by the COL applicant, or a completely new QA program, the staff will perform the necessary reviews in ensure compliance to 10 CFR Part 50, Appendix B. This was DFSER COL Action Item 17.1.1-1.

17.2 Quality Assurance During the Operations Phase

The operations QA program is beyond the scope of GE's application for design certification (DC) and was identified by the staff in the DFSER as DFSER COL Action Item 17.2-1. This item was addressed by GE Amendment 31 of the SSAR (SSAR Section 17.0.1.1), which is acceptable to the staff. GE has also included this action item in the SSAR. The adequacy and acceptability of the SSAR is evaluated in Chapter 1 of this report.

17.3 Reliability Assurance Program During Design Phase

Introduction

ABWR SSAR Section 17.3 describes the reliability assurance program (RAP) for the design phase of the ABWR. GE implements the design reliability assurance program (D-RAP) for its scope of design during detailed design and specific equipment selection phases to ensure that the important ABWR reliability assumptions of the probabilistic risk assessment (PRA) will be considered throughout plant life. The COL applicant will augment and implement the D-RAP for its scope of design and equipment selection (DFSER COL Action Item 17.3.1-1). Additionally, the COL applicant will develop and implement an operational reliability assurance program (O-RAP) which monitors equipment performance and evaluates equipment reliability to provide reasonable assurance that the plant is operated and maintained commensurate with PRA assumptions so that the overall safety is not unknowingly degraded and remains within acceptable limits (DFSER COL Action Item 17.3.9-1). When structures, systems, and components (SSCs) monitoring and evaluation identifies performance or condition problems, appropriate corrective action will be taken to assure SSCs remain capable of performing their intended functions. However, the RAP does not attempt to statistically verify the numeric values used in the PRA through performance monitoring.

The staff has evaluated SSAR Chapter 17.3, "Reliability Assurance Program During The Design Phase," which included the GE response (dated March 5, 1992) to the staff's request for additional information (RAI) contained in its request for resolution of issues related to SSAR Chapter 19, Appendix K

Open Item F17.1.3-1 *Inspection of QA Program*

Insert G, SER page 17-5

An inspection of the ABWR design process was performed from September 7 through 10, 1993. The inspection results are documented in NRC inspection report 99900403/93-02. The inspection scope included an examination of GE QA controls applied to the ABWR project. This included a review of design record files (DRFs), selected computer codes used for accident analysis and transient modeling, test activities, design calculations, and audits. The inspection questioned the technical adequacy of supporting calculations generated by the international Technical Associates (TAs). Some test data for the Full Integral Simulation Test could not be retrieved by GE, and some calculation notebooks were poorly maintained. The staff evaluation of GE's response to the findings of that inspection was Open Item F17.1.3-1.

GE provided a response to the staff's inspection report on November 24, 1993, which addressed the items of concern and proposed corrective and preventive actions such as: verifying the accuracy of an input parameter for a LOCA analysis and performing related sensitivity studies, disseminating training reminders to technical staff about the QA requirements for design analysis and DRFs, increasing the GE audit emphasis on the content of DRFs, verifying that installed test instrumentation was within specified tolerances, supplementing transient analysis code DRFs, confirming that engineering services were provided under the auspices of an Appendix B quality program, correcting SSAR inaccuracies, and performing design verification on a design calculation. The staff found these proposed actions to be acceptable with a few exceptions. A request for further information and clarification was sent to GE on December 22, 1993, for the issues involving the technical oversight by GE of supporting calculations generated by the TAs and the conduct of computer code design verification. GE's response dated January 17, 1994, was found to be acceptable with one exception discussed as follows.

During the course of the inspection in September 1993, the staff identified that the common engineering documents (design specifications, process flow diagrams (PFDs), instrument block diagrams (IBDs), and piping and instrument diagrams (P&IDs)) have received a considerable level of GE design review. However, the level of GE review performed on the supporting calculations generated by the international TAs was not found to be as rigorous. For example, the NRC inspection found that the depth of technical review afforded by the GE program reviews (QA audits) was minimal as the audit teams had not been supplemented by technical reviewers. In addition, little documented evidence was found in the DRFs to substantiate GE's review of the supporting calculations.

GE informed the staff that a sufficient level of confidence was obtained in the supporting calculations through the performance of GE program reviews of each TA, the GE engineering reviews of the common engineering documents, and participation by GE staff in numerous design review meetings. In addition, GE provided amplifying information during meetings with the staff on March 14 and 15, 1994, with respect to the extensive GE involvement during the ABWR design evolution. GE stated that, during the period from 1978 through 1985, extensive technical interaction transpired between GE and the TAs.

On March 22 through 24, 1994, a second NRC inspection was performed to substantiate the extent of the GE technical oversight of the TA's supporting design and analysis efforts. The inspection spanned a representative sampling of ABWR systems for which a TA had lead design responsibility. The staff examined the associated GE DRFs, interviewed cognizant GE design engineers, reviewed engineering correspondence from the TAs, and searched for examples of GE verification of TA calculations.

The three-day inspection resulted in the identification of evidence of GE's technical oversight of the supporting design as documented by the Phase 3 "Advanced BWR Plant Evaluation Report", GE comparisons of the ABWR design parameters with respect to the BWR 5 and 6 plant designs, thorough GE review of the common engineering documents that included proposed design revisions and independent GE calculations, the existence of selected TA supporting calculations in the GE DRFs, and GE review of system analysis, system performance, and capacity calculations generated by the TAs.

The inspection determined that reasonable assurance was provided by the depth, extent, and duration of the GE technical oversight of the joint design process to resolve the remaining issue from the September 1993, inspection. During the March 1994, inspection the staff additionally reviewed selected GE corrective and preventive actions that had been implemented in response to other concerns raised during the September 1993, inspection and found them satisfactory. Therefore, Open Item F17.1.3-1 is resolved based on the March 1994, inspection findings and the corrective and preventive measures instituted by GE in response to the QA and design control concerns identified in NRC inspection report 99900403/93-02.

Enclosure 2

Open Item F6.2.1.9-1

Description of Item	Suppression pool strainer
Advance SER Page No.	6-29
Insert No.	N/A
Attachment:	Letter of March 30, 1994, to GE

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

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

All ECCS suction

6.2.2 Containment Heat Removal System

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

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

- Low-Pressure Flooder (LPFL) Mode

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

- Suppression Pool Cooling Mode

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

The sizing of each strainer should consider all LOCA scenarios for which that system impacts the design basis or the probabilistic safety assessment (PSA) risk, or is relied upon within the EOPs.

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

March 30, 1994

Docket No. 52-001

Mr. Joseph F. Quirk
GE Nuclear Energy
175 Curtner Avenue, MC-782
San Jose, California 95125

Dear Mr. Quirk:

SUBJECT: REMAINING ISSUES ON ADVANCED BOILING WATER REACTOR (ABWR) REVIEW

In a letter dated March 9, 1994, the staff provided the Advisory Committee on Reactor Safeguards (ACRS) with revisions to the advance copy of the safety evaluation report (SER) on the ABWR design that addressed 13 of the remaining issues. An issue that was not addressed in that letter involves quality assurance for the ABWR design. The staff is currently preparing the revision to the SER for that issue based upon our recent inspection.

The purpose of this letter is to advise you that the staff has reconsidered its position on Open Item F6.2.1.9-1, Suppression Pool Strainer Size, as set forth in the March 9, 1994, letter. The staff has recently completed its reevaluation of GE's proposed resolution in its letter dated February 14, 1994, and determined that GE has not adequately addressed this issue. The staff's position on this issue is enclosed. I also want to remind you that the missing combined license action item for an updated probabilistic risk assessment, as discussed in Open Item F1.9-1, needs to be included in your standard safety analysis report (SSAR).

The current status of these remaining issues will be addressed in a letter to the ACRS prior to its April full committee meeting. If you have any questions or desire further discussion on these issues, please contact Son Ninh at (301) 504-1125 or Dave Tang at (301) 504-1147.

Sincerely,

R. W. Borchardt

R. W. Borchardt, Director
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

ADVANCED BOILING WATER REACTOR (ABWR) EMERGENCY
CORE COOLING SYSTEM (ECCS) SUCTION STRAINERS

The technical staff has reassessed the potential impact of clogging of the ECCS suction strainers on the GE ABWR design. The staff has considered GE's position which required that the DHR strainers be sized at three (3) times the area determined according to the method referenced in Regulatory Guide (RG) 1.82, Revision 1, for all loss-of-coolant accidents (LOCAs) except the main steam line (MSL) and reactor core isolation cooling (RCIC) steam line breaks. For those breaks, the strainers must be at least equivalent to the area calculated according to the RG. The GE position allows the high pressure core flooder (HPCF) and RCIC strainers to be sized according to the RG, without the factor of three enhancement.

The staff has conducted a qualitative assessment of the risk associated with not applying the three-times multiplier to (1) the steam line breaks for the decay heat removal (DHR) system and (2) the design of the RCIC and HPCF strainers. The risk analysis shows that the incremental risk is marginal, unless very pessimistic assumptions are used.

Nevertheless, there remain uncertainties in our knowledge of the severity of this phenomenon on the design basis of the ECCS. Recent technical assessments for operating reactors have led the staff to issue NRC Bulletin 93-02, Supplement 1, which requests interim compensatory actions to minimize the potential for loss of ECCS suction pressure as a result of a LOCA. Further analysis is required to assess the impact of non-fibrous debris on the potential for head loss. The staff has not yet bounded the magnitude of this issue.

In light of these uncertainties, and considering the limited impact that this issue could have on the cost of constructing an ABWR, it seems prudent to consider a more conservative position, to ensure compliance with 10 CFR 50.46 regardless of the outcome of the ongoing research program. This approach is in line with the agency's goal of providing a greater margin of safety for next-generation reactor designs.

Within the ECCS design basis, the high pressure systems (HPCF and RCIC) are not credited for long term recirculation and core cooling. However, these systems are options available to the operators and included within the Emergency Operating Procedures (EOP) for response to all accidents, including breaks in the MSL. Accordingly, the staff believes that the suction strainers for these systems should be sized for the spectrum of breaks for which they would be relied upon within the EOPs.

The staff believes that the concerns expressed above should be addressed by requiring that <u>all</u> ECCS suction strainers be sized to three times the area that would be calculated based on RG 1.82, Revision 1. The sizing of each strainer should consider all LOCA scenarios for which that system impacts the design basis or the probabilistic safety assessment (PSA) risk, or is relied upon within the EOPs.

Enclosure