July 27, 2009

ORGANIZATION: GE Hitachi Nuclear Energy (GEH)

PROJECT: Economic Simplified Boiling Water Reactor (ESBWR) Design Certification

SUBJECT: SUMMARY OF THE AUDIT OF THE PROBABILISTIC RISK ASSESSMENT SUPPORTING CHAPTER 19 OF THE ESBWR DESIGN CERTIFICATION APPLICATION

From May 6-8, 2009, the U.S. Nuclear Regulatory Commission (NRC) staff conducted an audit of the Probabilistic Risk Assessment (PRA) which supports Chapter 19 of the ESBWR Design Certification Application. The audit was conducted at the GEH offices in New Hanover, North Carolina. An audit summary, including participants, audit activities, a summary of the exit meeting, and a list of requests for additional information (RAIs) issued, is provided in Enclosure 1.

/RA/

Dennis J. Galvin, Project Manager
ESBWR/ABWR Projects Branch 1
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-010

Enclosures:
1. Audit Summary

cc: See next page

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1. Audit Summary
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Nuclear Regulatory Commission Staff Audit of Economic Simplified Boiling Water Reactor Probabilistic Risk Assessment Supporting Chapter 19 of the Economic Simplified Boiling Water Reactor Design Certification Application
Audit Summary

1.0 INTRODUCTION

The Nuclear Regulatory Commission (NRC) staff conducted an audit of the Economic Simplified Boiling Water Reactor (ESBWR) Probabilistic Risk Assessment (PRA) which supports Chapter 19 of the ESBWR Design Certification Application. The audit was conducted at the GE Hitachi (GEH) Nuclear Energy ("the applicant") offices in New Hanover, North Carolina during the period May 6 – 8, 2009. The staff conducted the audit in accordance with NRC Office of New Reactors (NRO) Office Instruction NRO-REG-108 [1]. The plan for this audit is documented in a memorandum from Hossein Hamzehee (USNRC) to Jeffrey Cruz (USNRC) dated April 14, 2009. [2]

2.0 PARTICIPANTS

The following NRC staff members from the NRO, Division of Safety Systems and Risk Assessment (DSRA) and Division of New Reactor Licensing (DNRL) participated in the audit:

- Mark Caruso (Audit Team Leader)
- Donald Dube (DSRA Senior Level Advisor for PRA)
- Edward Fuller (Senior Risk & Reliability Engineer)
- Hossein Hamzehee (Chief for PRA Licensing, Operational Support, and Maintenance Branch 2)
- John Lai (Risk & Reliability Engineer)
- David Misenhimer (Project Manager)
- Marie Pohida (Senior Risk & Reliability Engineer)

The following NRC contractors from Energy Research Incorporated participated in the audit:

- Roy Karimi
- Moshen Khatib-Rahbar

The following key individuals from the applicant's organization participated in the audit on a regular basis:

- Richard Wachowiak (Nuclear Island Engineering Manager)
- Gary L. Miller (PRA, Severe Accident Technical Lead)
- Lou Lanese (ESBWR Design Control Document (DCD) Chapter 19 Regulatory Affairs Manager)
- Glen Seeman (PRA engineer)
- Jonathan Li (PRA engineer)
- Eric Browne (PRA engineer)
- Jill Watson (PRA engineer)
- Yan Gao (PRA engineer)
- Lynn Crawford (Senior Staff Administrator)
- Lovisa Wallin (PRA contractor)
- Bill Berg (PRA Engineer)

3.0 AUDIT ACTIVITIES

The audit was conducted by a team of NRC staff and contractor personnel knowledgeable in the ESBWR PRA. The audit covered the full range of topics addressed in the staff's safety evaluation report. A summary for each of the topics covered is provided below.

3.1 Initiating Events

3.1.1 Identification of Initiating Events

The staff's principal goal in this review was to confirm that the applicant's process for identifying initiating events was capable of identifying ESBWR design-specific initiating events. The staff questioned the applicant regarding the process used to identify initiating events for the ESBWR and reviewed the description of the process in Chapter 2 of draft NEDO-33201, Revision 4 [3]. The staff found that a systematic and thorough process was used to identify initiating events for the ESBWR design. The staff found that the applicant's process includes: (1) an engineering review of each initiating event identified historically for other Boiling Water Reactor (BWR) product lines, (2) a system-by-system review of ESBWR systems with the potential for causing initiating events, and (3) consideration of human errors that could cause initiating events given the current state of knowledge regarding operations, maintenance and testing activities expected from personnel at a ESBWR site. The analysis for individual ESBWR systems is discussed in Section 2.2.3 of draft NEDO-33201, Revision 4 [3], entitled "Special Initiators". In performing these analyses, the applicant looked specifically for system failures that could result in reactor scram and/ or system isolation. The effects of spurious equipment actuation, human errors, hardware failures and pipe breaks in these systems were also considered. The applicant explained that the purpose of these evaluations was not only to identify initiating events but to also identify the impacts of those events on other systems in the ESBWR. In this regard, the staff examined the applicant's modeling of the loss of air systems initiating event, which is discussed in Section 3.1.2 below.

The staff found that the applicant's approach for identifying initiating events is consistent with applicable high level and supporting requirements for Capability Category II in the current American Society of Mechanical Engineers (ASME) PRA Standard [4]. The staff confirmed that the applicant's process is capable of identifying a reasonably complete set of initiating events for the ESBWR.

3.1.2 Treating the Impact of Initiating Events on Accident Sequences

To investigate the applicant's methods for treating the impact of initiating events on individual system performance during accidents, the staff examined the treatment of the loss of instrument air event. The staff discussed the treatment with the applicant and reviewed applicable system fault trees and event trees in draft NEDO-33201, Revision 4 [3]. The staff found that initiator impact in event sequences is handled in the following way. First, the loss of instrument air initiating event and other support system failures that initiate accident sequences are included as events in the system fault trees for systems that are affected by the events. Special designators are used to indicate that they are initiating events and not basic events. Event trees are then quantified by first recasting them as large fault trees and modeling each core damage sequence with core damage as a top event with an AND gate that includes the initiator as well as the top events for each of the applicable system failures in the sequence.

In the specific case of loss of instrument air, the staff found that because this event always leads directly to a loss of feedwater transient in the ESBWR, the event trees for the loss of air initiator and loss of feedwater initiator look the same. Consequently, for purposes of computational efficiency, a single large fault tree is used to treat both initiators. The initiator in the large fault tree is modeled as an OR-gate, i.e., loss of feedwater OR loss of instrument air. The methodology for quantifying the large fault trees includes logic that prevents double counting the same initiator in cut sets and removes cut sets that include both initiators. The applicant's methodology is an adequate means for capturing the impact of an initiator on mitigating systems in the cutsets for an accident sequence.

3.1.3 Categorization of Initiating Events

It is stated in Chapter 22 of NEDO-33201, Revision 3 [5] that a loss-of-coolant accident (LOCA) caused by a rupture in the Standby Liquid Control System (SLCS) injection line had been recategorized from a medium liquid LOCA to a small liquid LOCA. The staff reviewed the bases for this change as part of the audit.

The staff found that Section 2.2.3.8 of NEDO-33201, Revision 3 [5] had been revised to explain the reasons for the change in SLCS LOCA event categorization. In this section of the report, the applicant explains that a re-analysis of the event using the SLCS line nozzle throat size rather than the pipe size to calculate the flow area and accounting for the actual routing of the line resulted in a LOCA event which met the criteria for a small LOCA. Small liquid LOCA's have a break flowrate which is less than the high pressure injection make-up capacity and require depressurization for effective low pressure injection. The staff reviewed the results of the applicant's SLCS LOCA analysis and confirmed that the initiating event has these characteristics.

3.2 Data Analysis

As part of the site audit, the staff reviewed Chapter 5 of the draft NEDO-33201, Revision 4 [3], "Data Analysis," and carried out a number of discussions on related reliability data issues with the applicant's staff. The general approach for developing the component reliability database was reviewed, which included the component failures, common cause failures, and data uncertainty. The staff reviewed a sample of basic events and their associated failure data to ensure that the approach was appropriately applied and final values were reasonable. Since ESBWR is still in the design stage and no, or limited, operating data is available, the applicant used generic reliability data for similar components and equipment. The generic data sources used for the ESBWR components included NUREG/CR-4550 [6], Advanced Light Water Reactor (ALWR) Utility Requirements Document (URD) [7], and EGG-SSRE-8875 [8], which were all found to be appropriate.

In addition, the staff performed a more detailed review of the failure data associated with the containment vacuum breakers, squib valves and the digital Instrumentation and Control (DI&C) system components. Concerns regarding the adequacy of this data were raised by the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee for Review of ESBWR Design Certification in a meeting with the staff on August 21, 2008 [9]. Specifically, the staff reviewed the following:

- The staff reviewed the prior failure distribution and the resulting failure probability for containment vacuum breakers and found the values to be reasonable. However, the staff noted that it was not clearly described in the report as to how these values were obtained.

Similarly, it was not clear how the failure probability values for squib valves were estimated. The applicant agreed to provide additional clarifications in Chapter 5 of NEDO-33201, Revision 4.

- The staff reviewed failure probability estimates for digital trip modules and common cause failure of digital trip modules. The staff acknowledged that there is insufficient data and operating experience related to the Digital Instrumental & Control system. The values used in the applicant's PRA report seem to be reasonable for gaining risk insights. However, these values need to be updated once more design information and operating data become available. The staff noted that the technical basis for these failure probability estimates was not provided in the report. The applicant agreed to describe how these values were obtained in Chapter 5 of NEDO-33201, Revision 4 [3].

The staff does not plan to identify any new request for additional information (RAIs) related to the data analysis task as a result of this audit, and there are currently no open items associated with this review area. However, based on the results of the NRC audit, the applicant agreed to provide additional clarifications in Chapter 5 of NEDO-33201, Revision 4 to include the technical basis for the failure probability estimates used for vacuum breakers, squib valves, and DI&C components, as discussed above.

3.3 Accident Sequence Analysis

Prior to the audit visit of May 6-8, 2009, staff reviewed the accident sequence analysis, including all event tree changes as described in Section 22.3.2 of NEDO-33201, Revision 3 [5]. Based on the at-power internal initiating event contributions to core damage frequency as tabulated in Table 17.2-1 of NEDO-33201, Revision 3 [5], staff chose the following two events for detailed review during the audit:

- Loss of feedwater
- Inadvertent opening of a Relief Valve (IORV).

Additionally, top events R1 (control rod drive (CRD) late injection) and the equivalent R2 for anticipated transient without scram (ATWS) events in the loss of feedwater tree (Fig. 22.3-19 of NEDO-33201, Revision 3 [5]) and loss of feedwater ATWS tree (Fig. 22.3-2 of NEDO-33201, Revision 3 [5]), respectively, were reviewed to ensure that the events were incorporated properly.

At the audit, staff met with the applicant's staff and discussed the success and failure logic for all branch points in the most recently updated loss of feedwater tree. Included in this review was a detailed discussion of the basis for success. The staff found the event tree logic and structure to be proper, including the incorporation of top event R1. Based on this review, staff reviewed the IORV tree and found the differences in structure from the loss of feedwater tree to be logical and appropriate. The staff found the incorporation of top event R2 in the loss of feedwater ATWS tree, in a manner analogous to R1, to be proper.

3.4 Success Criteria

Prior to the audit, staff reviewed the success criteria as described in Section 3.3 of NEDO-33201, Revision 3 [5] and tabulated in Table 3.3-1. The staff also reviewed the changes to the success criteria as described in Section 22.3.2.2 of NEDO-33201, Revision 3 [5].

Based on the staff's review of the accident sequences discussed above as well the basis for success in Table 3.3-1, the staff chose the following top events for detailed review:

- VM: fire protection system (FPS) make-up (1/1 dedicated FPS reactor pressure vessel (RPV) injection pump and associated valves)
- VI: gravity driven cooling system (GDCS) injection (2/8 lines and 1/3 GDCS pools for loss of feedwater)
- UD: CRD for RPV injection (1/2 CRD pumps)
- WV: containment venting (vent path established).

The success criteria for all four of these top events are based on thermal-hydraulic analysis with the Modular Accident Analysis Program (MAAP).

For VM, the applicant provided the design flow rate for the dedicated pump. The staff found the pump to be designed for the appropriate flow rate, with margin, at an appropriately assumed back-pressure for the RPV and containment. The staff found the MAAP analysis to be appropriate in demonstrating acceptable success criterion.

For VI, the applicant provided the MAAP analysis for the loss of preferred (offsite) power event, which closely resembles if not bounds the loss of feedwater. The analysis shows the reactor vessel water level to briefly drop below the top of the active fuel, but there is no temperature excursion. The staff concluded the analysis to be reasonable and appropriate. Additionally, staff reviewed the basis for changes to the GDCS success criteria as described in 22.3.2.2.2 and found the basis to be adequate.

For UD, the applicant provided the nominal flow rate for one and two CRD pumps at normal reactor coolant system (RCS) pressure. The MAAP analysis showed that for the limiting case of a feed line break outside containment, the fuel is not uncovered. Additionally, staff performed a scoping analysis of the flow necessary to remove decay heat and confirmed the adequacy of the flow from one CRD pump.

For WV, the applicant provided the MAAP analysis results summary for the loss of preferred power event using low pressure injection. The nominally-sized 2-inch vent line is found to have insufficient capacity. The nominal 12-inch vent line is found to be more than adequate. Current design is for a nominal 14-inch line, which provides substantial margin. The staff found the analysis to be reasonable and appropriate.

3.5 System Modeling

In response to previous ACRS comments, the staff reviewed the ESBWR documentation regarding compliance with the system modeling requirements of the ASME/ANS PRA Standard [4], specifically, SY-A12 through A14. These supporting requirements are regarding the incorporation of passive and active failures of equipment that would affect system operability, including various flow diversion paths, failure modes, and numerical criteria for excluding very low contributors.

The applicant described the overall process that was applied to the system analyses, including the use of system templates that specify the events to model and assumptions.

Section 4.0.3 of NEDO-33201, Revision 3 [5] provides some of the assumptions of the system analyses, but the staff found that the guidance does not specifically address which failure modes in SY-A13 to consider and numerical criteria for exclusion per SY-A14. However, the applicant performed a self-assessment in November 2007 against the (then) ASME PRA Standard [4].

This self-assessment did address SY-A12 through A14 on a system-by-system basis, with observations and recommendations.

The staff reviewed the revised GDCS fault tree in draft NEDO-33201, Revision 4 in response to a comment by some ACRS members that certain important failure modes appeared to have been left out of the system analysis. The staff found that the fault tree has been revised to incorporate additional failure modes. For example, the failure mode plugs/transfers closed is included for maintenance valve F001A and maintenance valve F004A (common to RPV injection and containment deluge path). The staff selected the SLCS and Fuel and Auxiliary Pools Cooling System (FAPCS) fault trees for inspection and found the revised SLC system fault tree now includes maintenance valves plug/transfer closed on each train. For FAPCS, inclusion of plugs/transfers closed as a possible failure mode resulted in an important risk insight. This failure mode for one valve, in particular, was so critical that the valve position is now to be monitored and alarmed in the control room.

The staff reviewed the applicant's process for incorporating test/maintenance unavailability in system models. Treatment of test/maintenance unavailability is discussed in Chapter 4 of draft NEDO-33201, Revision 4 [3]. This chapter was not compete at the time of the audit and was not reviewed during the audit. Through discussions with the applicant, the staff found that in draft NEDO-33201, Revision 4 [3], assumptions regarding test and maintenance unavailability for each individual system model were generally based on the "best estimate" of treatment expected by the system engineer given the design of the system and relevant operating experience. In cases where "best estimate" expectations indicated no test and maintenance unavailability during power operation for systems covered by technical specifications, the applicant assumed a test and maintenance unavailability based on allowed outage time specified in the technical specifications. The staff considers such an approach to be consistent with applicable high level and supporting requirements for Capability Category II in the current ASME PRA Standard [4], and therefore acceptable.

3.6 Changes to the PRA (Revision 3 to Revision 4)

3.6.1 PRA Changes Resulting from Design Changes

The staff reviewed each of the documented changes to the ESBWR design since completion of NEDO-33201, Revision 3, including the applicant's documented assessment of how each change impacted the PRA. The staff confirmed that the only design changes affecting the PRA were those made to ensure that drywell pressure limits are not exceeded during design basis LOCAs.

These changes included the addition of logic to terminate feedwater addition following a LOCA, the addition of logic to isolate control rod drive make-up injection following successful operation of the GDCS and addition of a diverse alternate injection path for the CRDS in the event of a common mode failure of the ECCS, including use of an instrumentation platform diverse from that serving the isolation logic. The staff reviewed descriptions of the design changes, drawings showing the type and arrangement of hardware, changes to the CRDS system model and the new event trees for LOCA sequences. The staff also discussed these changes with the applicant's technical staff.

The staff also reviewed the revised text in Section 4.3 of draft NEDO-33201, Revision 4 [3] that describes the CRDS system model. The staff found that the CRDS design changes were being reflected in the fault trees of the system model appropriately, and that the logic in the LOCA event trees was sound.

3.7 External Events

3.7.1 Implementation of RAI Responses

One of the objectives of the audit was to verify that RAI responses regarding the sensitivity analysis for Internal Fire, Internal Flooding and strong winds had been incorporated into NEDO-33201, Revision 3 [5]. The applicant stated that additional design changes had been made (See Section 3.6.1) after submitting the RAI responses. Therefore, the results will be updated again and will be provided in NEDO-33201, Revision 4 [3]. The staff will confirm that the RAI responses have been reflected in NEDO-33201, Revision 4 [3] when it is complete and been made available to the NRC.

3.7.2 Internal Fire PRA Methodology

The applicant discussed its implementation of the Internal Fire PRA methodology referenced in the DCD with the staff. The following elements of the methodology were discussed in regard to application of the methodology to the full power operating condition.

- Plant Boundary and Partitioning
- Component Selection
- Fire PRA Cable Selection
- Qualitative Screening
- Fire Ignition Frequency
- Quantitative Screening
- Detailed Circuit Failure Analysis and Cable Selection
- Post-Fire Human Reliability Analysis (HRA)
- Seismic-Fire Interactions
- Fire PRA Documentation

The staff confirmed that the applicant has followed the methodology described in NUREG/CR-6850 [10] as completely and practically as possible, given that a facility has not yet been constructed. In addition, the staff confirmed that the updated fire PRA model included most recent ESBWR design changes and that these changes did not impact the fire PRA results significantly.

3.7.3 Seismic Margin Analysis

The staff has reviewed the approach that the applicant used for the PRA-based Seismic Margin Analysis. The staff reviewed the seismic event tree used to generate the High Confidence of Low Probability of Failure (HCLPF) values for the full power condition. The staff was able to trace the fault trees developed for the top events in the event tree and verified the results of the nodal HCLPF equations.

3.7.4 “Other” External Events and External Flooding

“Other” external events such as aircraft impact, industry accidents, pipeline accidents, hydrogen storage failures, vehicular and rail accidents were evaluated using the sensitivity analysis approach. The staff discussed the sensitivity analyses with the applicant and confirmed that the results show that the contribution to risk from these events are insignificant compared to other external and internal events, and these events can be screened out in the PRA analysis. The treatment of aircraft impact in the DCD will be revised after the NRC Aircraft Assessment rule [11] becomes effective.

For external flooding, the applicant stated that based on Regulatory Guide 1.200 [11], if the site meets the external flooding design criteria, the event can be screened out. The staff agrees with this position. The staff reviewed the following documents and was able to confirm that external flooding design criteria have been met.

- Draft ESBWR Revision 6 Tier 1, Section 2.16.5 “Reactor Building” and Section 2.16.6 “Control Building”
- Draft ESBWR Revision 6 Tier 2, Chapter 2, “Site Characteristics”, and Section 3.4.1.2, “Flood Protection from External Sources”.

3.8 Shutdown Risk Analysis

In response to staff RAIs, the applicant expanded the shutdown internal events PRA from NEDO-33201, Revision 3 to include operator induced reactor vessel drain down events and reactor water clean-up (RWCU) system breaks outside containment. The applicant also excluded credit for the non-safety related isolation of reactor water clean-up/shutdown cooling (RWCU/SDC) during Modes 5 and 6, since it is not required to be operable by Technical Specifications (TS) and is not included in the Availability Controls Manual. Therefore, as part of the site audit, the staff briefly reviewed each shutdown internal event tree and focused on the eight new event trees associated with reactor vessel drain downs and RWCU breaks outside containment. Four event trees were developed to evaluate reactor vessel drain downs and RWCU breaks outside containment in plant operational states (POSs), Mode 5 and Mode 5-open (event trees numbers 16.4-27, 16.4-28, 16.4-31, and 16.4.32). It is assumed that the reactor vessel head is on during these POSs. The staff found that these event trees credit successful GDCS operation without first checking for successful ADS via four DPVs. Successful opening of four DPVs is necessary for successful GDCS operation. This issue was identified as a follow-up activity during the audit. At the audit exit meeting, the applicant committed to correcting these event trees and re-quantifying the PRA model.

In addition to event tree review, the initiating event frequencies and the fault trees for these new event trees were evaluated. The initiating event frequencies were based on data in EPRI Technical Report 1003113 [13]. Screening values were assigned for failure of the operator to terminate leak paths. The staff found this approach to be acceptable. The staff reviewed the fault tree for failure to isolate breaks outside containment (top event BC-TOPRWCU) and found them to be complete and logical.

The core damage frequency due to high winds during shutdown modes in draft NEDO-33201, Revision 4 [3] increased by a factor of thirty from the high winds estimate reported in NEDO-33201, Revision 3 [5]. The applicant increased the hurricane initiating event frequency by a factor of five to account for coastal plant sites and improved the loss of offsite power modeling to account for initiator impacts on SSCs. The staff reviewed the revised high wind cutsets, Risk Achievement Worth (RAW) values, and structural capacities of SSCs.

The staff also reviewed the loss of offsite power event trees for shutdown modes and the fault tree associated with loss of power from all power sources (i.e., the unit auxiliary transformer, reserve auxiliary transformer and the non-safety related diesel generators). The staff considers the change in core damage frequency to be reasonable given the changes to the PRA.

The core damage frequency due to fire during shutdown modes in draft NEDO-33201, Revision 4 decreased over sixty percent compared to the frequency estimate in NEDO-33201, Revision 3. The applicant added a fire barrier such that both plant service water trains were housed in separate fire areas in the Service Water/Water Treatment Building. The new fire barrier is discussed in DCD Tier 2 , Section 9A.4.9 but was not reflected in DCD Tier 2, Revision 5 [14], Figure 9A.2-33 as stated in response to RAI 19.1-174S01. The applicant said that the drawing would be revised prior to issuance of Revision 6 of the DCD. The staff will confirm that the drawing has been revised appropriately when Revision 6 of the DCD is submitted.

RAI response 19.1.144 S03 was discussed with the applicant at the audit. This RAI pertains to isolation condenser (IC) operational issues during cold shutdown conditions. This RAI sought information on the operational impact of the RCS head vent, and the risk impact of isolation condenser system (ICS) unavailability for an indefinite time period (as allowed by TS). Discussions with the applicant resulted in a new supplemental RAI 19.1.144 S04 which was issued in a letter to the applicant dated May 14, 2009 [15].

3.9 Level 2 PRA and Severe Accident Evaluation

The Level 2 PRA and Severe Accident Evaluation audit focused on material that will be contained in Sections 8, 9, 10, and 21 of NEDO-33201, Revision 4 [3]. Draft versions of Sections 9 and 10 and new material that will be included in Sections 8 and 21 was reviewed. Particular attention was paid to documentation and RAI responses associated with two unresolved issues: ability to cool debris in the lower drywell utilizing the basemat melt arrest and cooling (BiMAC) device; and preparation of the technical basis for ESBWR severe accident management guidelines (SAMG).

3.9.1 BiMAC Device Testing Program

There were a number of open issues associated with the review of the applicant's topical report on the BiMAC device testing program, NEDE-33392P, Revision 0 [16]. These issues were reflected in RAI 19.2-95 S01, RAI 19.2-100 S01, RAI 19.2-104 S01, RAI 19.2-112 S01, RAI 19.2-124, and 19.2-125. The staff reviewed the responses to these RAIs and found that all responses except for the response to RAI 19.2-95 S01 were acceptable. At the audit exit meeting, the applicant agreed to provide a supplemental response to RAI 19.2-95 S01.

During the audit the staff raised a concern regarding the effects of non-condensable gas liberated as a result of interaction between corium and the ½ meter of sacrificial concrete on the BiMAC device. Such gas could adversely affect the performance of the Passive Containment Cooling System (PCCS). To resolve this concern, the staff has issued RAI 19.2-127, which requests that the applicant provide an analysis of the effects of the erosion of sacrificial concrete on the PCCS and containment performance.

3.9.2 Severe Accident Management

The staff reviewed Revision 1 of the Boiling Water Reactor (BWR) BWR Owner's Group Accident Management Guidelines Overview Document [17] document. Special attention was devoted to a proposed change to Section 4 of the document. This change includes the addition of actions to assure that the lower drywell is flooded and debris is cooled during hypothesized severe accidents in the ESBWR.

As a result of this review, the staff requested that the applicant revise the Inspections, Tests, Analysis and Acceptance Criteria (ITAAC) in Tier 1 Table 3.3-1 and the text in Tier 2 Section 14.3.3 to assure that the technical basis for severe accident management is utilized in preparing procedures and training modules. A new RAI 14.3-453 was issued in a letter to the applicant dated May 19, 2009 [18] to address this issue.

4.0 SUMMARY OF EXIT MEETING

The staff conducted an exit meeting with the applicant on May 8, 2009. The following individuals, in addition to those listed in Section 2.0, participated in the meeting:

- David Hinds, New Units Engineering Manager, GE-Hitachi
- Larry Tucker, ESBWR Engineering Manager, GE-Hitachi
- David Piepmeyer, Project Manager, ESBWR DCD, GE-Hitachi

The staff summarized its activities during the audit and discussed the following key preliminary results:

- Several open items regarding shutdown PRA were resolved.
- Several open items regarding severe accident analysis and accident management were resolved.
- Open items regarding High Winds analysis issues were resolved.
- Several new issues regarding the material change in the BiMAC device design were identified.
- Open issues regarding treatment of external flooding and other traditionally minor external events were resolved.
- Modeling errors were identified in four event trees used to evaluate reactor vessel drain downs and RWCU breaks outside containment in several shutdown operational states.

The staff identified the following follow-up activities based on the audit:

- The applicant agreed to provide additional clarifications in Chapter 5 of NEDO-33201 to include the technical basis for the failure probability estimates used for vacuum breakers, squib valves, and DI&C components.
- Due to incompleteness in results of fire PRA sensitivities, the results will be reviewed by the staff when NEDO-33201, Revision 4 is submitted to NRC.
- The treatment of aircraft impact in the DCD will be revised after the NRC Aircraft Assessment rule becomes effective.
- The applicant stated that drawing 9A.4.9 would be revised prior to issuing Revision 6 of the DCD. The staff will confirm that the drawing has been revised appropriately when Revision 6 of the DCD is submitted.

- The staff will confirm that the applicant has incorporated the RAI responses related to BiMAC and debris coolability, as necessary, into Chapter 21 of NEDO-33201, Revision 4 [3] and/or Chapter 19 of the DCD Revision 6 when they are submitted.
- The staff will confirm that the applicant has changed the ITAAC related to severe accident management, as committed, and added a COL item assuring that the technical basis for severe accident management would be utilized in preparing procedures and training modules.
- The staff will confirm that the applicant has corrected event trees used to evaluate reactor vessel drain downs and RWCU breaks outside containment and re-quantified the PRA model.

The staff made the following comments regarding the BiMAC device at the audit exit meeting:

1. Changes in the BIMAC device design, to the extent that are not proprietary should be reflected in the DCD Chapter 19 and the PRA sections, as appropriate.

2. The impact of design changes should be addressed by modifying the PRA and the DCD as appropriate.

The applicant stated that the issues involving the material change in the BiMAC device design would be addressed following receipt of a formal RAI that documents the staff's questions. The applicant also stated that modeling errors identified in event trees of the Shutdown PRA would be corrected, PRA model would be re-quantified and the results would be incorporated into NEDO-33201, Revision 4. The applicant stated that NEDO-33201, Revision 4 would be completed and submitted to the NRC by the end of June 2009.

5.0 RAIs Issued

The staff has issued the following RAIs based on the results of the audit:

- RAI 19.2-127 requests that the applicant provide an analysis of the effects of the erosion of sacrificial concrete on the PCCS and containment performance (See Section 3.9.1 above).
- RAI 19.1.144S04 requests confirmatory information regarding credit taken for operation of the Isolation Condenser in the shutdown PRA.

6.0 REFERENCES

1. Nuclear Regulatory Commission Office of New Reactors (NRO) Office Instruction, NRO-EG-108, USNRC, April 2, 2009.

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