ORGANIZATION: GE Nuclear Energy
San Jose, California

REPORT NO.: 99900403/94-01

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NUCLEAR INDUSTRY ACTIVITY: GE Nuclear Energy (GE) is engaged in the supply of Advanced Boiling Water Reactor (ABWR) designs worldwide. GE also furnishes engineering services, nuclear replacement parts, and dedication services for commercial grade electrical and mechanical equipment.

INSPECTION CONDUCTED: March 22 through 24, 1994

TEAM LEADER: Richard P McIntyre 5/31/94
Richard P. McIntyre Date
Reactive Inspection Section No. 1
Vendor Inspection Branch (RVIB)

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APPROVED: Uldis Potapovs 6-1-94
Uldis Potapovs, Chief Date
Reactive Inspection Section No. 1
Vendor Inspection Branch (VIB)

INSPECTION BASES: 10 CFR Part 50, Appendix B and 10 CFR Part 21

INSPECTION SCOPE: To determine the adequacy of the GE design oversight of their International Technical Associates (TAs), Hitachi and Toshiba, for the design and analyses efforts supporting ABWR design development.

PLANT SITE APPLICABILITY: None

1 INSPECTION SUMMARY

1.1 Violations

No violations were identified during this inspection.

1.2 Nonconformances

No nonconformances were identified during this inspection.

2 STATUS OF PREVIOUS INSPECTION FINDINGS

2.1 (Open) Nonconformance 99900403/(93-02-01)

Not reviewed during this inspection.

2.2 (Closed) Nonconformance 99900403/(93-02-02)

Nonconformance 93-02-02 stated that contrary to Criterion III of Appendix B to 10 CFR Part 50 and Section 4.4.1 of Engineering Operating Procedure (EOP) 40-3.00, "Engineering Computer Programs" (ECPs), the flow area of the internal recirculation pump used for the modeling of the SAFER code was based on an unverified hand drawn sketch with a reference to an individual who provided the information, instead of a reference to the applicable dimensioned design drawing.

The specification in question was the reactor internal pump flow area. GE used a flow area of 1.597 square feet in its original ABWR SAFER loss of coolant accident (LOCA) analysis. The inspection team spoke with GE staff who described the sensitivity analysis and the information added to the Design Record File (DRF). Specifically GE ran its code to include an area of one square foot and three square feet. The results of the three runs show that the minimum vessel water level and the peak clad temperature remain essentially unchanged with the different pump internal areas. In addition, GE included a current Japanese detailed design drawing for the reactor internal pump. This drawing indicates that the internal area of the pump is exactly 2.17 square feet. Based on the information provided, this nonconformance is considered closed.

2.3 (Open) Nonconformance 99900403/(93-02-03)

Not reviewed during this inspection.

2.4 (Open) Nonconformance 99900403/(93-02-04)

Nonconformance 93-02-04 stated that contrary to Criterion XII of Appendix B to 10 CFR Part 50 and Sections 1.1 and 4.2.b of EOP 35-3.20, "Calibration Control," GE purchased thermocouples used in the ABWR Full Integral System Test (FIST) tests from a commercial grade supplier, not on GE's approved supplier list, and accepted and used the instruments as calibrated by the supplier without further verification of the quality or traceability of those calibrations.

GE's response to this nonconformance, dated November 24, 1993, indicated that a detailed justification for the calibration of the thermocouples used in the FIST test would be placed in the DRF for the system. The staff reviewed the document and determined that the rationalization for the lack of pre-test calibration was a conscious decision which was not documented in the DRF. The actual "in-situ" calibration which was conducted before, during, and after the test was equivalent or more effective in promoting valid results. This information is acceptable as placed in the file. Regarding the second corrective action discussion, GE indicated in its response that other testing conducted since the FIST program has been properly documented to reflect the calibration of the instrumentation used. Discussions with GE staff revealed that GE had not conducted a detailed review of post-FIST testing to substantiate that statement, but only had talked with GE's calibration organizations's staff to obtain a general understanding, rather than a technical basis about the extent of calibration conducted in-house. This nonconformance is still considered open.

2.5 (Open) Nonconformance 99900403/(93-02-05)

Not reviewed during this inspection.

2.6 (Closed) Nonconformance 99900403/(93-02-06)

Nonconformance 93-02-06 stated that contrary to Criterion V of Appendix B to 10 CFR Part 50 and GE-NE SSAR for the ABWR, Chapter 17, Section 17.1.2, "Quality Assurance Program," GE failed to perform an annual implementation review of Hitachi and Toshiba's QA program for the 1991 period. This failure resulted in a 16 month interval between the audits performed in 1990 and the 1992 audits.

GE's response stated that the 1991 annual QA program implementation review was scheduled for November of that year. However, in October 1991, GE learned that the ABWR customer in Japan had planned an audit of GE at five different sites in the United States in early December. This audit required extensive preparation and meetings during November by the same GE personnel involved with the QA review and, consequently, the QA review was rescheduled to February, 1992 by mutual consent of GE and its associates. The response also stated that GE is implementing the commitment to perform annual audits on a calendar year basis rather than a 12-month interval and will perform annual implementation reviews through 1995. If future reviews go beyond the annual commitment, a note will be placed in the QA review file explaining the reason for the reschedule.

Records of the 1992 QA program review were examined during this inspection. The reviews were conducted in accordance with the revised schedule by a team of two GE auditors. Although some observations were discussed in the letter transmitting the review summary to the international technical associates, no corrective action requests resulted from these reviews. Based on the information provided, this nonconformance is considered closed.

2.7 (Closed) Nonconformance 99900403/(93-02-07)

Nonconformance 93-02-07 stated that contrary to Criteria V and VII of Appendix B to 10 CFR Part 50 and Section 7 of the GE QA Program Topical Report, NEDO-11209-04A, "Control of Purchased Material, Equipment and Services," GE failed to perform audits of Bechtel's ABWR QA Program Plan implementation for engineering services associated with GE PO No. 190-ALWR-31387, and accepted safety-related services from Bechtel without Bechtel being listed on GE's Approved Suppliers List for such services.

GE's response stated that Bechtel provided a letter to GE dated September 10, 1993, which confirmed that all Bechtel work supporting safety related items in the ABWR SSAR are in compliance with the project Nuclear Quality Assurance Plan and Bechtel Nuclear Quality Assurance Manual and that GE followed up with an audit of Bechtel to verify that the applicable quality requirements had been implemented. GE also stated that it reviewed all other purchase orders on the DOE contract under which this work was performed and confirmed that all other subcontractors were properly qualified. GE believes this nonconformance to be an isolated case caused by the separation of government business from commercial nuclear work and has since been corrected by the consolidation of the Approved Suppliers List for all the related business segments.

Records of GE's audit of Bechtel were reviewed during this inspection. The audit was performed by one individual on November 11, 1993 and included in its scope Bechtel's QA audits, design calculations and design verification activities. No nonconformances were identified by GE. Based on the information provided, this nonconformance is considered closed.

2.8 (Closed) Unresolved Item 99900403/(93-02-08)

Unresolved Item 93-02-08 stated that GE has the following statement in Chapter 17 of the ABWR SSAR: "The lead responsibility to produce each specification and drawing is formally assigned to one design organization. However, the content of each document is reviewed and approved by GE. While all common engineering documents reflect the formal consensus of all parties, GE is responsible for the design and supporting calculations and records for the ABWR project." The ABWR system DRFs did have the Japanese plant (K6/K7) system design specification, process flow diagram (PFD), piping and instrument diagram (P&ID), and instrument block diagram (IBD) for each system. These received a formal GE review via Engineering Review Memoranda (ERM) and the resolution of comments was well documented. However, there was a scarcity of information on supporting calculations, particularly for those systems in which the International Technical Associates (TAs), Hitachi and Toshiba, had the lead design responsibility. GE had not documented a review of the supporting calculations for the reactor building cooling water system and the audit process of the TAs did not examine the technical adequacy of the supporting calculations. The inspection questioned the technical oversight by GE of supporting calculations generated by the TAs and how GE met their Chapter 17 SSAR commitment.

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. During the period from 1978 through 1985, GE stated that extensive technical interaction transpired between GE and the TAs.

The inspection was performed to substantiate the extent of the GE technical oversight of the TAs 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 inspection resulted in the identification of evidence of GE's technical oversight of the supporting design as documented by the Phase III Advanced BWR Plant Definition Evaluation Final Report (Phase III Report), GE involvement in the Phase II design effort, 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 system 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 consider this unresolved item to be closed. Section 3 of this inspection report provides more detail as to the background and information reviewed by the inspection team.

2.9 (Closed) Unresolved Item 99900403/(93-02-09)

The GE SSAR had contained two inconsistencies with respect to the ABWR design information. GE deleted the erroneous process flow diagram for the reactor building cooling water system from the SSAR. That level of design detail was not needed by the NRC staff to reach a safety conclusion on the system for design certification purposes. GE additionally rectified the value for the main steam flow rate in Table 6.3-1. The team reviewed the conduct of GE verification for a sample of changes, Engineering Change Notices (ECN) 00951 and ECN 0093, that GE plans on incorporating into the SSAR. Each planned change was accompanied with a detailed review sheet documenting the accuracy of the information to be included in the SSAR.

Based on the correction of the specific examples of erroneous information contained in the SSAR, and the conduct of an on-going GE verification of future SSAR amendments, this unresolved item is considered closed.

2.10 (Open) Unresolved Item 99900403/(93-02-10)

The team examined some corrective action aspect associated with this item. Audit S&PQ 93-17 was conducted between the period November 15, 1993, to January 7, 1994. The audit scope included the DRFs for several systems. One

Corrective Action Request (CAR) resulted from the audit with regards to the identification of additional problems with respect to design verification conduct. An additional training session was conducted by GE to indoctrinate the engineers as to the management expectations on the conduct of design verification. The GE corrective actions to the NRC item concluded that a single engineer was responsible for the concerns. During the course of this inspection, several DRFs under that engineer's cognizance were examined and the trend of informal design practices continued to be identified. The team was presented with a closed out CAR from the S&PQ 93-17 audit. Because of time constraints the team was unable to verify the adequacy of GE's actions pertaining to this unresolved item so it will remain open.

3 INSPECTION FINDINGS AND OTHER COMMENTS

3.1 Background:

An inspection of the ABWR design process was performed from September 7 through 10, 1993, to evaluate the effectiveness of the GE quality assurance (QA) program implementation. 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 system DRFs, selected computer codes used for accident analysis and transient modeling, test activities, design calculations, and audits. During the course of the inspection, the team identified that the common engineering documents such as design specifications, PFDs, IBDs, and P&IDs had received a considerable level of GE design review. However, the level of GE review performed on the supporting calculations generated by Hitachi and Toshiba was not as rigorous. The inspection questioned the technical adequacy of supporting calculations generated by Hitachi and Toshiba.

GE provided a response to the staff's inspection report on November 24, 1993, which addressed the nonconformances and unresolved items with proposed corrective and preventive actions which the staff found to be acceptable with a few exceptions. A request for further information and clarification was sent to GE on December 22, 1993. GE's response dated January 17, 1994, was found to be acceptable with the exception of Unresolved Item (93-02-08). 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. However, the NRC inspection found 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 existed in the GE DRFs as to the substance of the design review meetings.

In discussions with the staff, GE suggested that the staff consider the amount of interaction, review, and oversight that GE exercised during Design Phase II (1981-1983) and Phase III (1984-1985), and the Common Engineering review which began in 1986 after the conclusion of the Phase III design effort. GE indicated that this was a period of intensive communications and design development between GE and the TAs which was a consensus process during which three initial designs for each system evolved into a single acceptable final design based on input from the technical staff of GE and the TAs. GE further stated that by reviewing the Plant Definition Evaluation Phase II and III final reports summarizing the joint design evolution effort, as well as GE system DRFs, the staff could potentially find sufficient evidence to support a positive conclusion relative to the unresolved item.

Prior to the inspection, the staff requested GE to review its ABWR system records and identify the systems for which a TA had lead design responsibility. From that set, the staff chose eight systems and during the inspection, the staff reviewed DRF documents to ascertain to what extent GE had reviewed specific design information and supporting analysis. Further, the staff searched for examples of GE verification of TA calculations. In addition, the inspection team members interviewed several GE staff engineers who previously participated directly in, or were cognizant of, the above activities. The systems reviewed by the team are discussed below.

3.2 Reactor Water Cleanup System

Hitachi had lead responsibility for the design of the reactor water cleanup (CUW) system during the Common Engineering review for the Japanese K6/K7 plants. GE had lead responsibility for this system during Phase III of ABWR design development. The SSAR and the Certified Design Material (CDM) for the ABWR design submitted for NRC certification were based on the K6/K7 design.

The team reviewed DRFs G33-0051-2, G33-0051-3 and G31-0028 for the CUW system to assess the extent of GE's review of the K6/K7 design information that is included in the SSAR and CDM, and to verify that the sources of selected design data presented in the CDM and SSAR are documented in the DRFs. The team did not review any calculation for correctness of methodology, reasonableness of the results or accuracy. Additionally, the Phase III Report was reviewed for this system as well as all other systems reviewed. The SSAR or CDM values for system capacity, heat exchanger sizes, and certain physical parameters were chosen for review.

The CUW system capacity of 2% of the rated feedwater flow was based on a 1974 GE study report which recommended this capacity instead of the 1% of feedwater flow rate used in BWRs to reduce the time required for restoration of reactor coolant water chemistry following plant shutdown. The DRFs contained correspondence between GE and its partners on the cost benefit analysis of specifying CUW system capacity of 1%, 2%, and 4% of feedwater flow rate and the reasons for selecting 2% as the basis for the ABWR.

The acceptance criteria in the CDM ITAAC for the centerline of the vessel bottom head drain line tee connection is 460 mm above the center line of the variable leg nozzle of the RPV wide-range water level instrument. This requirement was based on engineering judgement that the operator could control the water level at about 15" above the top of active fuel in the event of a break in the CUW system line outside the containment and the failure of the isolation valves to close. To resolve a comment from the NRC independent review group, GE had decided to delete this requirement from the CDM and the SSAR. The team found this approach acceptable.

The CDM and SSAR specify a maximum throat diameter of 135 mm for the flow restrictor in the CUW suction line. This was based on the differential pressure across the restrictor needed for accurate flow measurement. The team found that a conservative flow area based on 70% of the internal diameter of the 200 mm suction line was used in the mass and energy release calculations in the ABWR compartment pressurization analysis (DRF-T11-00007, Section 7) prepared in support of the SSAR. The calculation also used the 30 second isolation valve closing time specified in the CDM and SSAR. The team concluded that the GE calculation used crtical input values consistent with the CDM and SSAR.

The regenerative and non-regenerative heat exchanger sizes for K6/K7 were reviewed by GE as part of the review of the system design specification submitted by its partner. In addition, GE reviewed the heat exchanger data sheets. In response to comments from GE, the partner provided sample calculations for the heat exchanger sizes. GE's approval of these documents confirm GE's review and acceptance of the design data for the ABWR.

The team found in the DRFs evidence of GE's review of P&ID, PFD, design specification, equipment requirement data sheets, IBDs, and resolution of GE's comments on these documents. The team also found that the selected design data from the CDM and SSAR were either supported by GE calculations or were reviewed by GE if supplied by its partners. The team concluded that the bases for the design data presented in the CDM and SSAR for the CUW system exist in the DRFs, and that GE had taken actions to ensure the validity of the data.

3.3 Fuel Pool Cooling and Cleanup System

Toshiba had lead responsibility for the design of the fuel pool cooling and cleanup (FPC) system during the Common Engineering review for the Japanese K6/K7 plants. GE had the lead responsibility for this system during Phase III of ABWR design development. The CDM and SSAR for the ABWR submitted for NRC certification were based on K6/K7 design.

The team reviewed DRFs G21-00020-2, G21-00020-3, and G41-00026 for the FPC system to assess the extent of GE's review of K6/K7 design information, and to verify that the sources of selected design data presented in the CDM and SSAR are documented appropriately in the DRFs. The DRFs were reviewed to verify the basis for the FPC heat exchanger capacity. The FPC system heat exchanger capacity for the K6/K7 plant was independently calculated by GE and its partners and was discussed at a working group meeting held in January 1985. The agreed upon capacity per heat exchanger was further refined and was included in the heat exchanger performance requirement data sheet ST-28-0204, Rev. 1 / HX-002, Rev. 3, prepared for K6/K7. The heat exchanger data was reviewed by GE and included in the SSAR.

The calculation supporting the RHR-FPC Joint Heat Removal Performance Table (SSAR Table 9.1-12) was found in DRF G41-00020-3. This calculation was specifically performed by GE to respond to an NRC staff question regarding the maximum fuel pool temperature when the RHR heat exchangers are used to supplement FPC system during plant shutdown. The format and contents of this calculation pre-dated the guidelines in GE procedure EOP 42-1.00, and therefore was left to the discretion of the individual performing the calculation because no general procedure for format and content of calculations was in force prior to December 1992. The safety evaluation in the SSAR Subsection 9.1.3.3 implies that the maximum possible heat load was based on the decay heat from a full core plus decay heat from spent fuel discharged at previous refueling up to a maximum spent fuel storage capacity of 270% of a core. However, the calculation in support of SSAR Table 9.1-12 assumed the decay heat load specified in Branch Technical Position ASB 9-2 which is different from 270% of a core. GE agreed to add a note to the table in Amendment 34 of the SSAR to clarify the basis for the heat load assumed in calculations in support of Table 9.1-12.

The DRFs contained GE comments on the FPC system PFD, P&ID, design specification, equipment performance requirement data sheets, and interlocking block diagrams. Appropriate GE approvals were found in the final documents indicating GE's acceptance of resolution of its comments by the partners. The DRFs contained correspondence between the parties related to comment resolution. DRF G41-00026 also included copies of excerpts from its partner's DRF in Japanese on the FPC system relating to heat load summaries, PFD, and P&ID for K6/K7. The team concluded that the bases for the design data in the CDM and SSAR for the FPC system exist in the DRFs, and GE had taken actions to ensure the validity of the data by independent analyses, comparison with other BWRs or checking the data provided by its partners.

3.4 HVAC Emergency Cooling Water System

Hitachi had lead responsibility for the design of the HVAC emergency cooling water (HECW) system during the Common Engineering review and during Phase III of the ABWR design development for the Japanese K6/K7 plants. GE did not have any lead responsibility for this system during Phase III. The design information provided by Hitachi has been used in the CDM and SSAR with major changes by GE. The changes include addition of a third division to cool the diesel generator room, relocation of chillers and pumps to the control building, realignment of main control room cooling from divisions A and B to divisions B and C, and increase of cooling capacity to meet CDM site parameters.

The team reviewed design information contained in DRF P25-00001-1/-2/and -3. This DRF contained a calculation for sizing the pumps and piping based on revised heat loads. The K6/K7 design was for a two train system whereas for the ABWR design, the system was expanded to a three train configuration. The chillers and pumps were re-sized and the team was informed that the sizing calculations were in the associated HVAC system files. During the course of the design development, GE requested the TA to provide the design basis of the surge tank capacity calculations. The TA responded with a description of how losses from valves and pumps along with system expansion considerations resulted in the final tank sizing. GE also provided comments on design details including: cooling provided to the diesel generator spaces, and control room cooling under single failure situations (loss of a pump/chiller).

In response to a question regarding heat loads in SSAR Table 9.2-9, GE showed a calculation in DRF U41-00002 which revised the K6/K7 heat loads for the reactor and control buildings to account for the higher ambient conditions stated in the SSAR. The result of this calculation was consistent with the SSAR table.

The team found documentation in the DRFs of the HECW design specification, equipment requirement data sheets, P&ID, PFD, system studies, related ERM correspondence between GE and the TAs, and Phase III Report information. The team also found that selected parameters from the CDM or SSAR were either supported by GE design documentation or were reviewed by GE if supplied by design documents supported by a TA.

3.5 Standby Liquid Control System

Toshiba had lead responsibility for the design of the standby liquid control (SLC) system during the Common Engineering review for the Japanese K6/K7 plants. GE had the lead responsibility for this system during Phase III of the ABWR design development. The SSAR and the Certified Design Material (CDM) for the ABWR design certification were based on the K6/K7 design.

The team reviewed DRFs C41-00114 and C41-00106 for the SLC system to assess the extent that design data presented in the CDM and SSAR are documented in the DRF. The SSAR or CDM values for system injection flow, boron concentration in the SLC tank, and certain physical parameters were chosen for review.

The Phase III design calculations for the SLC system considered the volume of water in the reactor and shutdown cooling system piping at shutdown condition. The requisite concentration and delivery rate of the sodium pentaborate were computed with allowances for mixing variability. The system design was based on a GE design specification for the redundant reactivity control system from a previous BWR design. Toshiba provided detailed technical comments, backed up by alternate calculations, with respect to the system design envisioned initially by GE.

During the course of the ABWR design evolution, the TA made a decision to increase the requisite boron concentration from 800 ppm to 850 ppm. Documentation existed in the DRF to demonstrate that GE had reviewed the associated analysis and had concurred with the rationale for the change. Evidence of detailed GE technical oversight was also apparent with respect to the equipment requirement specification generated by the TA for SLC where GE questioned aspects associated with the pump design net positive suction head requirements. In response to GE's inquiry, the TA provided a hand calculation to GE for review.

The DRF yielded considerable insight into the GE technical review of the SLC system design specification, dated December 22, 1986. GE generated 44 comments on the specification that included technical questions with respect to: the stated boron concentration, the dilution due to the shutdown cooling system, the total time to achieve injection of the entire quantity of boron solution into the vessel, and the sizing of the tank heaters based on prior designs. In response to the GE questions, the TA provided GE with relevant analysis and calculations to validate the design parameters contained in the design specification.

The team found documentation in the DRFs of the SLC design specification, equipment requirement data sheets, Phase III design report, P&ID, process flow diagram, system studies, and related ERM correspondence between GE and the TAs. The team also found selected parameters from the CDM or SSAR were either supported by GE design documentation or were reviewed by GE if supplied by design documents generated by a TA.

3.6 Drywell Cooling System

Hitachi had lead responsibility for the design of the drywell cooling system (DCS) during the Common Engineering review and during Phase III of the ABWR design development for the Japanese K6/K7 plants. The SSAR and the CDM for the ABWR design certification were based on the K6/K7 design.

The team reviewed the DRF for the DCS to assess the extent that design data presented in the CDM and SSAR are documented in the DRF. The SSAR or CDM values for average and maximum local drywell temperatures, reactor building cooling water - cooling coil design parameters, heating ventilating and air conditioning normal cooling - cooling coil design parameters, drywell cooling fan capacity, drywell cooling system heat loads, and certain physical parameters were chosen for review.

During the course of the design, there were a number of technical discussions on the most effective manner to achieve cooling, whether by having cooling supply ducts or cooling return ducts. Analysis, testing, and operating plant experience were considered by GE and Hitachi when determining the most appropriate method for the system design. It was noted, however, that certain information was not contained in the DRF. For example, the DRF did not document the bases for area-specific heat loads used to calculate the drywell cooling system capacity. It can be assumed that since this information was not included in the ERM documents, it was not reviewed as a part of this process.

The design specification was found to be consistent with the SSAR information pertaining to average drywell temperature and localized temperature criteria. Equipment specification information was found consistent with SSAR Table 9.4-1 with respect to parameters such as cooling capacity, air and water inlet/outlet temperatures, and drywell cooling fan capacity. However, during the course of reviewing the SSAR with respect to the DRF design specification information, the team identified that the SSAR Table identified the sensible heat loads for the wet well air space was 0.282×10^6 kcal/hr. The associated analysis stated the proper location was actually the upper drywell piping area and the associated heat load was 0.255×10^6 kcal/hr. GE stated that it had identified this inconsistency and that it would be corrected in Amendment 34 of the SSAR.

The team found documentation in the DRFs of the drywell cooling system design specification, equipment requirement data sheets, Phase III design report, P&ID, process flow diagram, system studies, and related engineering review memoranda correspondence between GE and the TAs. The team also found selected parameters from the CDM or SSAR were either supported by GE design documentation or were reviewed by GE if supplied by design documents supported by a TA.

3.7 Containment Atmospheric Monitoring System

Hitachi had lead responsibility for the design of the containment atmospheric monitoring system (CAMS) during the Common Engineering review and during Phase III of the ABWR design development for the Japanese K6/K7 plants. The inspection team reviewed the design description information included in the Phase II report which basically remained unchanged in the Phase III report. It is very similar to the information included in the SSAR and in the CDM for the ABWR. The inspection team also discussed the system design with cognizant GE staff to ascertain the nature of GE's oversight of the design process and interaction with its partners. GE personnel indicated that the design of the system is very similar to that of a Mark 5 design and the system characteristics are dictated by NUREG-0737 requirements, and follow the guidance in Regulatory Guides (RG) 1.97 and 1.7.

In addition, GE included in Section 6.19.6 of the Phase II report a comparison of ABWR features to those of the BWR/5 design. It was indicated that the ABWR has a wider range of radiation monitors to meet RG 1.97 with sensors mounted inside the primary containment vessel for more effective monitoring. For the hydrogen and oxygen monitoring, the BWR/5 design uses a wet basis reading within containment, while the ABWR uses a dry sampling technique taken outside of containment.

The inspection team reviewed approximately 40 to 50 ERM's processed during 1989 related to the CAMS design. Many of GE's reviews resulted in the mere identification of typographical errors in the Hitachi documents. Some resulted in GE providing significant comments on the quality of the design documents, which resulted in Hitachi modifications to produce a more detailed and acceptable design. Specifically, on December 14, 1988, GE commented on the Task Analysis Report and identified deficiencies in six areas that were subsequently addressed to GE's satisfaction. Also, GE commented on February 10, 1989, concerning the Man Machine Interface Requirement Design Specification. Lastly, on December 26, 1989, GE provided detailed feedback to Hitachi on the D23/CAMS Hardware Software Design Specification that resulted in Hitachi revising the IBD.

The inspection team also reviewed a letter from GE to Toshiba and Hitachi dated January 15, 1988, which identified problems related to the Japanese versions of documents. Modifications were apparently made to the documents after the three-party agreement was made. GE discovered these inconsistencies, investigated them, and found that they were, in fact, minor oversights on Hitachi's part. The issue was addressed by stamping Japanese versions of documents provided to GE, which certified that the English translations were complete and correct. This review process identified GE's attention to detail in the review of design documents and design transmittal documents.

3.8 Flammability Control System

GE performed the initial design work for the flammability control system (FCS) and transferred the lead to Toshiba after the Phase III report was issued. The inspection team reviewed the Phase III report system description. In this document the system is described as being compris d of two mobile combiner units to be shared among 4 Japanese plants on the same site. The source terms assumed were based on those included in RG 1.7. The capacity indicated per combiner was listed as 150 scfm with only 1 combiner required to perform the designated function. The system would be manually ini iated by the operator after a LOCA signal was received. The Phase III report also reflected a comparison of the ABWR FCS to that of the BWR/5 design. Subsequent to Phase III, GE independently modified the FCS design to reflect the permanent installation of two independent recombiners.

Most of the team review for this system concentrated on the documents produced after the Phase III report when Toshiba had the design lead. A significant amount of documents included in the DRF (which was not well organized) traced the design analysis produced by Toshiba which justified the need for only two portable combiner units for the four plants on its reactor site.

Initially, Toshiba provided a copy of its recombiner analysis in January of 1988. It proposed to justify a slower production of oxygen and hydrogen after a LOCA and a reduction of the gamma absorption from 10% to 8%. These changes were variances from the guidance in RG 1.7. Based on the inspection team review of the DRF, it identified that on several occasions in 1988 GE indicated that Toshiba's analyses were not conservative enough and needed to be improved to justify its position. On May 6, 1988, GE provided a detailed review of Toshiba's analyses, highlighting the differences in GE's methodology versus its partner's. Finally, on June 27, 1988, GE transmitted its detailed analyses of the gamma ray absorption fractions for post LOCA shutdown conditions to Toshiba indicating that it would be realistic and conservative to delay the initiation of recombiners to approximately 40 hours after a LOCA without exceeding oxygen concentration limits. This provided closure for the disagreement on the design basis for the K6/K7 design of this system.

The DRF reflected a considerable amount of independent analysis on the part of GE for the Toshiba system design to ensure that the design basis for the system was consistent with NRC regulatory requirements.

3.9 Control Building Safety Related Equipment Area HVAC System

Toshiba had lead design responsibility for the control building safety related equipment area HVAC system from its inception. Toshiba sized the system to accommodate a two unit heat load. This was a conservative input for the ABWR design further developed by GE. For this reason, GE did not do an independent verification of the heat load analysis conducted by Toshiba. Subsequent to the Phase III design effort, GE modified the design of the system to accommodate a wider range of outside temperature than that assumed by the Japanese to meet U.S. siting assumptions.

The inspection team reviewed the DRF for this system and found it relatively limited. It included an incomplete Toshiba P&ID for the system, the Japanese design basis for the system, some GE calculations and a copy of GE feedback on the K6 and K7 P&ID. The inspection team reviewed the April 9, 1990, letter from GE to its partners which included markups of the Japanese P&IDs and feedback from the results of GE's fire hazards analysis. The letter also provided a list of room numbers with remarks identifying required changes and clarifications related to cooling system classification, the need for fire barriers, and the qualification of instrumentation. The inspection team also reviewed the June 26, 1989, TASC transmittal letter from GE to its partners which provided additional detailed comments resulting from a configuration review of the P&ID for the system. GE provided over 25 comments, many of which resulted in the modification of the P&ID by Toshiba.

The inspection team also reviewed a copy of bulk temperature checks performed in 1993 on the K6/K7 design and ABWR design to verify SSAR heat loads. The checks were very informal yet they were consistent and accurate. Although this was a Toshiba design and GE implemented system modifications to meet U.S. licensing requirements, GE still exercised a limited amount of oversight in its configuration review and feedback.

3.10 System Review Summary

The system review resulted in the identification of evidence of GE's technical oversight of the supporting design as documented by the Phase III Advanced BWR Plant Definition Evaluation Final Report and documentation in the system DRFs of 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, related ERM correspondence between GE and the TAs, the existence of selected TA supporting calculations, and GE review of system analysis, system performance, and system capacity calculations generated by the TAs. The team also found that selected parameters from the CDM or SSAR were either supported by GE design documentation or were reviewed by GE if supplied by design documents generated by a TA.

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