CERTIFIED BY:
Carlyle Michelson - 11/5/93

CERTIFIED

DATE ISSUED: 10/22/93

ACRS-2896

PDR 3/25/94

SUMMARY/MINUTES OF THE ACRS SUBCOMMITTEE MEETING ON THE ADVANCED BOILING WATER REACTORS (GE) SEPTEMBER 8, 1993 BETHESDA, MARYLAND

PURPOSE

The purpose of this meeting was to discuss the resolution of open issues in the ABWR standard safety analysis report (SSAR), review GE's response to ACRS concerns, and to evaluate how the ABWR design satisfies the requirements resulting from the resolution of certain unresolved safety issues (USIs) and generic safety issues (GSIs). In addition, the Subcommittee discussed the NRC staff's schedule for submittal of the final safety evaluation report (FSER). The meeting began at 8:30 a.m., adjourned at 7 p.m. and was held entirely in open session. A copy of the meeting agenda is attached. Dr. Medhat El-Zeftawy was the Designated Federal Official (DFO) for the meeting. No written comments or requests for time to make oral statements were received from members of the public.

ATTENDEES: The principal attendees were as follows:

ACRS

C. Michelson, Chairman
J. Carroll, Member
I. Catton, Member
P. Davis, Member
T. Kress, Member
W. Lindblad, Member
R. Seale, Member
C. Wylie, Member
M. El-Zeftawy, Cognizant Staff Engineer

GE

J. Power
J. Fox
A. Beard
B. Genetti

NRC

J. Wilson, NRR
C. Poslusny, NRR
C. Craig, NRR
M. Malloy, NRR
J. Segala, NRR
C. McCracken, NRR
R. Borchardt, NRR
G. Bagchi, NRR
T. Boyce, NRR
M. Hum, NRR
J. Guttmann, OCM/FR

Others

N. Fletcher, DOE
S. Franks, DOE
R. Lipinski, EG&G
S. Frantz, Newman & Holtzinger

CHAIRMAN'S OPENING REMARKS

In his opening remarks, Mr. Michelson congratulated GE representatives for producing a clearly formatted and more readable

SSAR document (Amendment 31). Mr. Michelson urged GE representatives and the NRC staff to concentrate on brief and concise answers unless the ACRS members requested more explanation. Mr. Michelson stated that he has prepared a list of questions as a result of a limited review of a few selected portions of the SSAR. This list has been sent to GE representatives and the NRC staff prior to this meeting. For completeness purposes, the list is as follows:

A. Divisional Barriers and Secondary Containment

1. Where are the divisional barriers inside and outside of secondary containment and in the reactor building? Fig. 9A.4-1, note 3, indicates colors for each division but the drawing is not in color. Note 4 talks about cross hatched colors but none are shown. Without these colors it is difficult to know where some of the divisional boundaries are located. How will the boundaries be accurately identified?

2. Concerning secondary containment and divisional boundaries, where can the structural design requirements such as design pressure, design temperature, hydrostatic loading, leak tightness, and resistance to pipe whip and jet impingement be found?

B. Main Steam Tunnel

1. The reactor building blowout panel is still shown on Fig. 1.2-2, 1.2-8, and 1.2-9. The main steam and feedwater line is incorrectly shown on Fig. 1.2-2. Section 3.4.1.1.2.2 states that the steam tunnel is sealed at the R/B end and open at the T/B end. Is the blowout panel in or out?

2. How large is the gap in the structural interface between the R/B and C/B tunnel and how is it sealed?

   Ditto for C/B to T/B joint? What internal tunnel pressure can the seals withstand?

3. If the structural joint is not sealed or the seal blows out during a steam line or feedwater line break, will the steam be drawn into the control building HVAC air intakes which are nearby and located in the confined space between the R/B and C/B?

4. Where are the temperature elements (TE) for the tunnel leak detection system located? They are not listed for Rm 440 (Table 9A.6-2) in the reactor building. Only TE020 A-D are shown in Fig. 5.2-8, Sheet 3, for the R/B. Why are there 9 sets of TEs shown for the turbine area and only one in the R/B?

5. What is the purpose of the steam tunnel vent shaft in the turbine building?

6. Section 3.4.1.1.2 refers to normally closed floor drains in the tunnel area. Where are the design and interface requirements prescribed? Are there valves in the floor drain piping? Where are they located? How are the lines routed to radwaste?

7. Is there a leaktightness requirement for the tunnel area? Where prescribed? (There are essential electrical panels directly below the steam tunnel area in the R/B.)

8. How is the tunnel HVAC equipment compartment sealed from the R/B secondary containment area? Is the HVAC compartment capable of remaining sealed when exposed to steam line break pressures?

C. Fire Protection

1. Fig. 1.2-4 shows a symbol for 3 hour fire rated floors but none are indicated in the figure. Why is the symbol shown if not used? Fig. 9A.4-3 shows that a number of floor areas are fire rated.

2. Since a water fog system is being specified for the diesel-generator compartments, should NFPA 11 be added to Table 1.8-21?

3. Concerning Table 1.8-21 (p. 1.8-38) will ASTM Fire Test Standards such as E119-88 (Standard Test Methods for Fire Tests of Building Construction and Materials) and E152-81a (Standard Methods of Fire Tests of Door Assemblies) be used and should they be included?

4. Where are the requirements stated for fire detection and mitigation under the false floor in the control room? How will the space be divided to meet physical separation requirements? Are there floor drains in the concrete floor below and how will they be routed to sumps? Will this area be normally ventilated to remove heat or is it passively cooled? How will it be ventilated for smoke removal during a fire? What is the estimated combustible fuel loading? Are there floor drains? Where is this false floor area discussed in the Fire Hazards Analysis? It does not appear to be included in 9A.4.2.4.1 (Control Room Complex). This

area appears to be the equivalent of a cable spreading area of unspecified dimensions and materials of construction included within the control room. More information is needed concerning the types of cable present, cable entry points, separation provisions, power loads, etc.

5. Section 3.4.1.1.2.1.4 states that for an RCW failure in the diesel-generator room, water will, "fill the floor area and escape into the corridor, with potential cascading down the stairwell." Is the compartment floor at El 12300 mm? Is it fully curbed? If as described in the quotation, a large fuel oil release to the room could escape and cause a potential fire to spread beyond the compartment.

D. Reactor Water Cleanup System

1. What is the final status of RWCU compartment design requirements for the pipe rupture case and what are the effects of a postulated RWCU pipe rupture on secondary containment and divisional boundary design?

2. With regard to the use of carbon steel piping for the RWCU system, does the present design consider the possibility of erosion/corrosion problems? What kind of experience is there with carbon steel piping in RWCU systems?

3. Will there be a nil ductility problem during a cold repressurization of the carbon steel? How will pipe ductility be assured?

4. Where are the test requirements concerning the ability of the RWCU containment isolation valves to interrupt flow under various postulated pipe rupture flow conditions?

Demonstrating the ability of a valve in an unbroken system to open or close under full differential pressure conditions may not be an adequate proof test of its pipe break flow isolation capability.

5. Has a pipe break analysis been performed for a postulated rupture of the RWCU return line to the feedwater system? The rupture of interest is downstream of check valve F014 which is probably within Rm 443. Unless adequately protected from the break, the normally open isolation valve F015 is assumed to be damaged by the event and remain open. The assumed single active component failure is for normally open check valve F006 at the return connection to the feedwater line to fail to close. The RWCU leak detection system may experience significant time delays in detecting the break. Since FE016 in the differential flow detection network is upstream of check valve E014, this network is not expected to detect the RWCU break. Fig. 5.2-8, Sheet 7, indicates that temperature elements TE009 A-M are provided to detect elevated room temperature from a break. table 9A.6-2 indicates that these elements are located in Rm 141 (E1-8200 mm), Rm 147 (E1 -8200 mm) and Rm 241 (E1 -1700 mm). The rupture location is likely to be in Rm 443 (E1 12300 mm) which has no TEs. It may take considerable time for the steam released by the hot water break to reach these temperature elements in the lower reaches of the building. Given enough time, the break may be detected by the main steam tunnel temperature detectors. Reactor water level should remain essentially unchanged since the feedwater flow diversion to the break will be relatively small compared to the total feedwater flow. The small change is accommodated by the feedwater control. Given enough time, the flow to the break will be terminated by the loss of feedwater system inventory.

6. It should be noted that the absence of TEs for break detection in Room 443 may also introduce significant time delays in the detection of breaks downstream of containment isolation valve F003 on the supply line to the RWCU. Is it acceptable not to have temperature leak detection in Rm 443 for this break?

7. Section 9A.4.1.4.20 indicates that Rm 443 contains no safety-related equipment, but Table 9A.6-2 shows the room to contain a number of important safety-related components including the containment isolation valves (F003 and F017). Can both references be correct?

E. Miscellaneous

1. Fig. 8.3-1, Sheet 1, show Div. I, II, and III being normally fed from the unit auxiliary transformer. In San Jose we were told that one division was going to be fed from the reserve auxiliary transformer in order to avoid a common mode overvoltage or regulator failure problem. Where will this requirement be specified?

2. Concerning section 1C.2.5.10, what site-related weather conditions will the CTG building be protected against? Will weather-related missiles be included?

3. Concerning section 1C.2.5.7, the CTG is to recharge the plant batteries during the SBO, how will battery room ventilation be provided?

4. For Fig. 9.2-7, Note 2, the battery rooms and associated electrical equipment rooms should also be protected. Where is the gum lined pipe shown in this figure described?

5. Why are the RIP motor generators located directly over the control room? Is there any missile problem? What are the other hazards (e.g., fire and fire mitigation) associated with placing such high energy electrical components so close to the main control room?

6. How are the backup control room cooled? How do the applicable HVAC normal and chilled water sources connect back to the UHS? What AC and DC power sources are required?

7. Concerning all essential HVAC chillers, how will they be restarted after a momentary power interruption, a loss of offsite power, or after an SBO? Is it a requirement that the compressor oil be kept heated with battery or CTG power for the SBO case?

8. Does brittle fracture of carbon steel chilled water piping need to be considered for seismic qualification? What requirements are specified to assure ductility at low temperature?

9. Where are the power supply assignments for the ADS valve shown?

Flood Design (Section 3.4)

1. Table 3.4-1 indicates a S/B to R/B access way at El 4800 mm. It is not clear where this access appears on the R/B or S/B drawings.

2. The table does not list the S/B or C/B access way at El 7900 mm (Fig. 1.2.19). Is it designed for flood protection?

3. The table indicates a S/B to T/B access way to E1 7900 mm but it is not shown in Fig. 1.2-19. Which is correct?

4. What is the S/B to T/B access shown at E1 3500 mm on Fig. 1.2-18? (Note that T/B does not go to this depth.)

5. The S/B to R/B access shown at E1 3500 mm on Fig. 1.2-18 cannot ben found on the R/B drawings. Is it designed for flood protection?

6. The table indicates a radwaste building pipe tunnel from the R/B and T/B to be at E1 1500 mm which is very deep. The T/B does not go below E1 5300 mm (Fig. 1.2-28). The R/B interface with the tunnel cannot be found on the R/B drawings. Where is this tunnel located? Is it the same as the radwaste tunnel which is at E1 8800 mm in the T/B?

7. Where are the RSW pipe chase and radwaste tunnel located?

8. How many chases and tunnels are there?

9. How are they sealed at building interfaces (e.g., T/B and R/B)?

10. Are the seals redundant at the safety-related building interfaces.

11. Are there seals at each end of the chase or tunnel?

12. How much hydrostatic pressure are the seals designed for?

13. Are the chases and tunnels seismically qualified?

14. Are they sealed against water intrusion during site floods?

15. Are there floor drains in each chase or tunnel?

16. How are internal pipe ruptures accommodated?

17. What other pipes or components such as control and/or electrical power cables are located within each chase or tunnel?

18. Are there other chases or tunnels connected to the C/B, R/B, or T/B (other than main steam)?

19. Are there any other penetrations other than door ways (e.g., pipe or electrical) and how are they sealed?

20. Where are the floor drain interface requirements prescribed for the R/B and C/B. Are there any extra requirements for floor drains inside of secondary containment? How are the divisional separation requirements for the floor drains assured?

    There may be special floor drain requirements for certain rooms such as those for the diesel-generator compartment and battery rooms.

21. What is the piping arrangement from R/B sumps to radwaste? Are there individual lines from each sump or are they headered with check valves in the branches for isolation?

22. Section 3.4.1.1.2.1.8 states that main reactor hall flooding on 4F drains into the service pools. Are there floor drains to the pool? Is there a high curb around the pool?

23. Section 3.4.1.1.2 appears to deal only with the effects of water on the floor. The water is generally confined to the room by curbs at the doors and a floor drain system. Safety-

related equipment is placed on pedestals. The section does not appear to deal with water sprays or leaks through the floor joints or penetrations such as for piping, cabling, or hatches. How will this and other forms of water migration be controlled by design requirements?

24. In section 3.4.1.1.2.1.1, what is the pressure retention capability requirement for the water tight doors, i.e., what is the maximum water elevation for which the doors will be designed? Will divisional separation walls (including penetrations) be water tight to the same elevation? Where is this requirement stated?

25. In section 3.4.1.1.2.2, are water tight floors provided above the control room? How will pipe ruptures or fire mitigation be accommodated without leaking water into the control room from above? The past good practice has been to place the control room above such large potential sources of water.

Codes, Standards and Guidance

1. Has the staff fully accepted all of the codes and standards listed in Table 1.8-21? How will any staff exceptions be identified?

2. In Table 1.8-20, pages 1.8-33, RG 1.143 is indicated as being in Table 17.0-1 but it is not there. conversely, RG 1.144 and 1.146 are in Table 17.0-1 (page 17.0-2) but not indicated as such in Table 1.8-20. How will this type of correction be handled?

3. If a COL applicant wishes to use piping purchased to ASME standards, what will be the acceptable rules for selecting the nearest size, etc. in English units? Do these rules appear in the SSAR?

COMPONENT CLASSIFICATION (Section 3.2)

1. What component classification scheme will be used for the ABWR, i.e., Table 3.2-1 or Fig. 1.7-1, Sheet 2, Note 11? How can they be made to be equivalent?

2. In Note 11, what are the requirements for NSC (non-seismic category 1)?

3. A number of drawings use the classification schemes given in Fig. 1.7-1. Will they be changed to reflect the requirements of Table 3.2-1 or will some kind of equivalency instruction be added to the SSAR?

4. Fig 9.4-1, Sheet 1, Note 1, for the C/B HVAC specifies a design/quality classification 7G. Ditto for 9.4-4, Sheet 1, Note 1, for the essential electrical equipment HVAC. Fig. 9.2-3, Sheet 1, Note 1, for the HVAC Emergency Cooling Water System shows group class 7C. What is the difference between the requirements for 7C and 7G in Fig. 1.7-1 and why do these essential HVAC and supporting systems have a NNS safety designation? Table 3.2-1 gives the safety-related equipment in these systems a safety class 3 designation.

5. Section 3.2.3.3 indicates that secondary containment is SC-3 and Table 3.2-1 indicates that the secondary containment valves and dampers are SC-2. What are the SC requirements for electrical and mechanical penetrations for secondary containment? Logically, this should also include doors and HVAC penetrations.

6. What are the design and construction requirements for the secondary containment walls that would be in keeping with their safety classification? Are they clearly specified in the SSAR?

NRC STAFF PRESENTATION

1. Tier 1, Tier 2, and ITAAC - J. Wilson, NRR
   Mr. Wilson stated that GE has applied for design certification of the ABWR under the provisions of 10 CFR Part 52. As endorsed by the NRC, the design certification process is proceeding on the basis of a tiered approach. Tier 1 will be the certified rule and will include a description of the principal design bases and principal design features of the certified design together with the ITAAC.

   Tier 1 material will include:

   - A design description together with the ITAAC entries for each of the approximately 100 systems in the ABWR facility for which design certification is being sought.

   - Proposed Tier 1 treatment for issues such as equipment qualification, radiation protection, and technical issues for which certification will be based on approval of design acceptance criteria (DAC).

- Interface requirements and the associated ITAAC as called for by 10 CFR Part 52 for those portions of the plant for which design certification is not being sought.

- A definition of the site-related parameters which have been used as input to the ABWR design process. These site-related design parameters have been selected with the intent they envelope conditions at most potential sites in the United States.

The Tier 2 material will encompass the larger body of design material submitted as part of the certification application as documented in the SSAR.

2. Closure of Open Issues and NRC Schedule - C. Poslusny, NRR

Mr. Poslusny stated that there were approximately 650 unresolved items identified as open and confirmatory in the draft SFR. By June 1993, the number had been reduced to approximately 46 "Punch List" items. Currently, less than 25 items remain as confirmatory and are being worked by the NRC staff and GE.

Mr. Poslusny indicated that the Tier 1/ITAAC review effort has started approximately a month ago with the staff identifying over 800 concerns. Currently, however, there are less than 20 issues remaining, with agreed upon GE actions. The staff is planning to verify that all resolutions are correctly and completely documented in GE's submittal of Amendment #32/SSAR (scheduled for mid-September 1993).

The NRC staff will provide preliminary FSER chapters to the ACRS as follows:

- September 28, 1993 - Chapters 2, 5, 8, 10, 11, and 12
- October 19, 1993 - Chapters 3, 4, 6, 7, 9, 15, and 17
- December 9, 1993 - Chapters 13, 14, 16, 18, 19, and 20
- December 17, 1993 - Chapters 1, and 22

USIs & GSIs

1. GE Presentation - Mr. B. Genneti

The ACRS selected the sample of USIs and GIs for detailed presentation by GE.

- USI/A-17: Systems Interactions in Nuclear Power Plants--As specified in NUREG-0933, this issue addresses the concern that inconspicuous or unanticipated interdependencies may exist between systems and may result in a degradation of the predicted capability of safety systems in an accident or transient, in particular from flooding and water intrusions.

  The acceptance criteria are the plant design should include the mitigation of internal flooding and water intrusion to preserve safe shutdown.

  Mr. Genetti stated that the ABWR is analyzed and evaluated to the extent practicable to reduce the risk from internal flooding and water intrusion. Building evaluations are discussed in Subsection 3.4 and indicate this issue is resolved for the ABWR.

- USI/A-36: Control of heavy loads near spent fuel--The concern is dropping heavy loads on spent fuel.

  The acceptance criteria are for the design to provide equipment, procedures and training to preclude the release of radioactivity for any credible heavy load drop.

  For the ABWR, the resolution is identified as follows:

  - The COL will establish heavy load routing,

  - Operating, maintenance, and training documentation is required.

  - Crane inspection and testing would be performed.

  - Critical equipments are designed to have single failure proof switches.

  - The design would have applicable safety factors.

  The relevant requirements are specified in Subsection 9.1.5 and 9.1.6. GE considers this issue resolved.

- USI/A-43: Containment emergency sump performance--The issue is:

  - Adverse RHR suction intake conditions and subsequent pump failure post-LOCA,

  - Transport of LOCA-generated debris to the RHR pump suction pool strainer to reduce the NPSH margin below required,

- Capability of RHR pump run with air or particulate ingestion on pump seal and bearing systems.

The acceptance criteria are for the design of the suppression pool and the RHR pump suction pool strainer should be adequate to assure long-term recirculation cooling flow capability post-LOCA.

For the resolution, the ABWR does the following:

- The design is resistant to the transport of debris to suppression pool,

- The SPCU system provides early indication of potential problems,

- RHR pump suction pool strainers are designed to meet the current regulatory requirements, and

- Equipment in drywell and wetwell minimize the potential for generation of debris.

GE considers this issue resolved for the ABWR design.

- USI/A-47: Safety Implications of Control Systems--This issue deals with the concern of the potential for accidents or transients (e.g., overpressure, overfilling, reactivity events) being made more severe as a result of control system failures including control and instrumentation power supply faults.

  The acceptance criteria are as follows:

- Plants shall provide automatic reactor vessel feedwater overfill protection. Tech. Spec. shall include provisions to verify.

- Design to minimize inadvertent trips of the main feedwater system.

For the resolution of this issue, the ABWR reactor vessel has level 8 overfill protection by stopping the feedwater pumps. The plant procedures will be developed by the COL applicant. Periodic testing is required by Tech. Specs. GE considers this issue resolved for the ABWR design.

- <u>GSI-43: Reliability of Air Systems</u>--The concern is the potential for nonsafety related air systems impacting air operated safety-related component performance. Safety-related equipment relying upon air systems to actuate or perform its intended function shall either fail safe upon loss of air or operate with the assistance of backup accumulator.

  The acceptance criteria for the resolution of this issue are that the design and operations of the instrument air system (IAs) shall be such that the reliability of the IAs is assured and that the IAs shall meet the intent of GL88-14.

  For the resolution, the ABWR does the following:

  - Instrument air quality is tested periodically

  - COL license information is included to ensure the provision of adequate maintenance practice, and

- Loss of instrument air is preoperationally tested.

GE considers this issue resolved for the ABWR design.

- GSI-87: Failure of HPCI Steam Line Without Isolation--This issue concerns a postulated break in the HPCI steam supply line and the uncertainty regarding the operability of the HPCI steam supply line isolation valves under the postulated conditions. A similar situation can occur in the RWCU system.

  The acceptance criteria to resolve this issue are defined in GL89-10, which requires adequately sited actuators for MOV's, verification, and a program for testing, inspection and maintenance to assure design basis performance.

  For resolution of this issue, the ABWR does not have an HPCI system. It does, however, have an RWCU and a RCIC. The ABWR addresses the concerns and issues identified in GL 89-10.

  GE considers compliance with GL 89-10 resolves this issue for the ABWR design.

2. NRC STAFF PRESENTATION - Ms. M. Malloy, NRR

Ms. Malloy presented a tally of USIs and GSIs to be evaluated for the ABWR. She stated that from Appendix B of NUREG-0933; there are:

- 28 USIs and task action plan items,
- 28 New generic issues,
- 4 Human factors issues,

- 91 TMI action plan items, and issues not included in Appendix B of NUREG-0933, there are 7 issues resolved without issuance of new requirements.

In total, there are 128 issues resolved with issuance of new requirements, 7 resolved without issuance of new requirements, and 23 issues without complete resolutions. These 23 issues are as follows:

- 10 High priority (B-56, 15, 23, 105, 121, 143, 153, HF4.4, HF5.1, HF5.2)

- 9 Medium priority (B-17, I.D.3, B-55, B-61, 57, 78, 89, 106, 142)

- 2 Possible resolutions identified [I.D.5(3), 83]

- 2 Resolutions available (145, 155.1)

As far as the specific issues discussed by GE above, for **USI/A-17**: the SSAR markup was transmitted to the staff in September 1993, and is being evaluated. **USI/A-36**: has been resolved. **USI/A-43**: discussion between staff and GE is underway. **USI/A-45**: being addressed by plant-specific severe accident analysis and PRA. **USI/A-47**: SSAR markup being transmitted and the staff is evaluating it. **GSI/43**: portions of this issue has been addressed in the ABWR design, with the remaining to be addressed by the COL applicant. **GSI/87** has been resolved.

<u>Interface Requirements</u> - A. James, GE

Mr. James described GE approach to comply with 10 CFR Part 52.47(a)(1)(vii). This regulation states that "the interface requirements to be met by those portions of the plant for which the application does not seek certification. These requirements must be sufficiently detailed to allow completion of the final safety analysis and design-specific probabilistic risk assessment required by paragraph (a)(1)(v) of this section."

Mr. James stated that GE's intent is to have a design certification application that includes interface requirements for out-of-scope (i.e., not part of the certified design and will be provided by license applicant on a site-specific basis) portions of the plant. The SSAR includes definition of interface requirements for all systems totally or partially out of scope. The Tier 1 material will include definition of principal interface requirements. No ITAAC for interface requirements.

Systems which are partially or totally outside the scope of the certified design are as follows:

System	Out-of-Scope
Ultimate heat sink (UHS)	Fully
Offsite power system	Partial
Makeup water preparation system	Fully
Potable and sanitary water system	Partial
Reactor service water system	Partial
Turbine service water system	Partial
Communication system	Partial
Site security	Fully
Circulating water system	Partial
Heating, venting and air conditioning	Partial

Interface requirements for out-of-scope systems (such as UHS) addressed in the SSAR system section are:

- Safety design bases (Temperature limits, heat loads, failure criteria, separation, testing, etc.)

- Power generation design bases (heat loads)

- Performance evaluation (Make-up limits, temperature limits, instruments and control, tests, inspections, etc.)

Mr. James stated that the interface requirements (IR)/SSAR entries are sufficient to allow completion of final safety analysis because major technical issues were addressed and extensive GE/NRC interactions have resulted in comprehensive IR that support the review.

In response to Mr. Michelson's question of how the IR can be verified through inspection, testing, or analyses, Mr. James stated that successful development of ITAACs for the certified design provides sufficient confidence that similar ITAACs can be written for the Tier 1 IR and the implementing site-specific designs.

In conclusion, Mr. James stated that the GE/ABWR SSAR and Tier 1 submittals are sufficient to meet the intent of Part 52.

Mr. Michelson asked if a COL applicant wishes to use piping purchased to ASME (i.e., English units) standards, what will be the acceptable rules for converting specified metric sizes to the nearest English equivalent sizes? Mr. James responded by stating that the SSAR identifies nominal pipe sizes in millimeters. The ABWR design work was performed using Japan industrial standards (JIS). The JIS and U.S. piping standard are essentially equivalent. The differences are sufficiently small that no re-engineering will be required for use of U.S. piping.

Divisional Barriers and Secondary Containment Issues - J. Power, GE

In response to Mr. Michelson's question of where can the structural design requirements such as design pressure, design temperature, hydrostatic loading, leak tightness, and resistance to pipe whip and jet impingement be found, Mr. Power stated that:

- The structural requirements for wall/compartments subject to High Energy Line Breaks (HELB) can be found in Appendix (3H.4) The primary spaces effected by HELB are the Reactor Water Cleanup and the Reactor Core isolation Cooling (RCIC). These walls are designed for 15 psid.

- The design basis for hydrostatic loading is applied to the divisional (flood) boundaries located on the basement levels of the reactor building and control building. The walls, doors, and penetrations are either designed for the worst case external hydrostatic load or located above the maximum flood water level. The divisional flood boundary walls in the basement of the reactor building are a minimum of 0.6 meters thick. This thickness is to prevent against seepage and is in excess of that required to resist the hydrostatic loads.

- The adequacy of the walls to resist jet impingement and pipe whip must be demonstrated by the COL since these items are dependent on the final piping layout and design.

- Leak tightness of the secondary containment boundaries will be determined by the Standby gas treatments ability to maintain the required negative differential pressure.

- Design temperatures for the various boundaries is bounded by the Environmental Qualification (EQ). The design temperatures of the boundaries in question equal or exceed those found in Appendix 3I. The short-term temperature spikes resulting from an HELB are not included.

<u>Main Steam (MS) Tunnel</u> - J. Power, GE

Mr. Michelson questioned (1) how large is the gap in the structural interface between the reactor building (R/B) and control building (C/B) tunnel and how is it sealed; (2) same question for C/B to turbine building (T/B) joint; and (3) what internal tunnel pressure can the seals withstand?

Mr. Power responded by stating that the steam tunnel joints between the R/B, C/B, and T/B will be constructed in an overlapping concentric sleeve arrangement. The overlap area will include a labyrinth arrangement to prevent radiation streaming. The clearances between the overlapping portions of the joints will have a 2-inch radial and axial clearance that (Refer to Figures 1.2.21 and 1.2-22) will be filled with an elastic/compressive material (e.g., RTV). A design pressure for the seals has not yet been determined.

MS tunnel concrete is designed with margin for approximately 15 psig.

<u>Fire Protection Issues</u>, J. Power, G.E.

Mr. Michelson asked, "Where are the requirements for fire detection and mitigation under the false floor in the control room? How will the space be divided to meet physical separation requirements? Are there floor drains in the concrete floor below and how will they be routed to sumps? Will this area be normally ventilated to remove heat or is it passively cooled? How will it be ventilated for smoke removal during a fire? What is the estimated combustible fuel loading? Where is this false floor area discussed in the Fire Hazards Analysis? It does not appear to be included in 9A.4.2.4.1 (Control Room Complex). This area appears to be the equivalent of a cable spreading area of unspecified dimensions and materials of construction included within the control room. More information is needed concerning the types of cable present, cable entry points, separation provisions, power loads, etc."

Mr. Power responded by stating the following:

- The raised floor area considered part of the Main Control Room and is included in the fire hazards analysis for fire area FC4910. The raised floor area will be used to route cable to and from the Safety System Logic Control (SSLC) cabinets, the operator bench boards and displays, and the divisional electrical equipment rooms.

- The control room area and the raised floor are considered to be non-hazard areas per IEEE 384. Section 8.3.3.6.2.2.3 discusses at length the separation criteria applied to divisional electrical cabling in the control room. GE has determined that fire detection and suppression equipment is not needed in the raised floor area. The justification for this position is based on the following:

  - The amount of cabling in this area is substantially reduced over current designs.

- The control room is continuously manned so that the presence of a fire will be quickly detected.

- The types of cables located in the raised floor area smolder for a long time and are usually self extinguishing.

- To the best of GE's knowledge, there has never been a fire in an operating plant that has required the evacuation of the control room.

- In the <u>extremely</u> unlikely event that the control room were to require evacuation, the remote shutdown panels provide the necessary controls to bring the plant to cold shutdown.

- The cabling that will be located in the raised floor area will be one of three types:

    - Fiber Optic Cables
    - Control and Signal Cables
    - Low Voltage Power Cables (<480 volts)

- Divisional separation of these cables will be maintained per the requirements of IEEE 384, Reg Guide 1.75, and GDC 17 (SSAR 8.3.3.1). For the raised floor area this effectively means that divisional cable trays will be separated by a minimum of 3 feet horizontally or will be enclosed with at least 1 inch clearance. Furthermore, all low voltage power cables will be contained in flexible or rigid conduit in the raised floor areas. Cables contained in conduit or enclosed trays are not considered to contribute to the combustible loading for the room.

- The divisional panels are physically separated as much as practical and located above the divisional electrical equipment rooms. The cabling from the divisional electrical equipment rooms will be routed to the (SSLC) cabinets with Divisions I and III on one side of the operator area and Divisions II and IV located on the opposite side of the operator area.

In closing, Mr. Michelson thanked GE representatives on their fine effort to support the Subcommittee meeting and to respond to ACRS concerns and noted that additional Subcommittee meetings will be scheduled to discuss the NRC staff's FSER regarding the ABWR design.

ACTIONS, AGREEMENTS, AND COMMITMENTS

1. Divisional separation--Barriers and design basis. Mr. Michelson asked GE to provide definition and detailed information relative to the divisional barriers inside and outside of secondary containment and in the reactor building. In addition, evaluate the divisional barriers performance under a spectrum of events and document the findings in the SSAR.

   GE representatives agreed to provide a written report responding to this request.

2. Radwaste tunnels--Design basis. Mr. Michelson asked GE to provide a description of the radwaste tunnels and evaluate their performance under a spectrum of events and document the finding in the SSAR.

   GE representatives agreed to provide a written report responding to this request.

3. Plant buildings--Flooding Protection (access door's design basis). Mr. Carroll asked GE to provide additional information relative to the flooding aspects through access/egress doors.

   GE representatives greed to provide such information with emphasis on prevention and mitigation of such event.

4. Nil ductility aspects of reactor water cleanup (RWCU) and HECW systems. Dr. Seale asked GE to provide additional information regarding the Nil ductility aspects for these two systems.

   GE representatives agreed to provide a written response.

5. USI/A-17--Systems interaction. Mr. Carroll asked GE to provide a comprehensive and complete response to how the ABWR addresses the USI/A-17 concerns and issues.

   GE representatives agreed to provide the requested additional information.

6. Safety system suppression pool strainer. Dr. Catton asked GE to describe and provide additional information regarding the ECCS/suppression pool strainer design basis.

   GE representatives agreed to provide the requested information.

7. Air Systems--Safety and non-safety components/FMEAS. Mr. Michelson asked GE to provide additional information relative to the performance of the air systems under adverse conditions and abnormal events and their impact on the plant safety functions.

GE representatives agreed to provide a brief review of air systems under adverse conditions and will evaluate their impact and performances on both safety and non-safety components.

8. Combustion Turbine Generator CTG--Design Basis and Tech. Spec. Requirements. Mr. Lindblad asked GE to describe the design basis of the CTG relative to site and building flood conditions.

   GE representatives agreed to provide the requested information.

9. Control Building - Floor drain and sump system (design basis). Mr. Michelson asked GE to provide additional information relative to flood/fire water control in the control building.

   GE agreed to provide the requested information.

10. Stainless Steel vs. Carbon Steel in RWCU System (Design Basis). Dr. Seale asked GE to provide additional information regarding this issue and also identify past uses and practices.

    GE agreed to provide the requested information.

Currently, GE is planning to submit its written responses to all of the above issues by the end of November 1993.

Future Actions

Future Subcommittee meetings will be scheduled as follows:

1. September 22-24, 1993 (Severe Accidents S/C): Re: ABWR Severe Accidents issues, MAAP code, and PRA considerations.

2. October 26-27, 1993 (ABWR S/C). Re: Review FSER Chapters 2, 5, 8, 10, 11, and 12.

3. November 2, 1993 (Computer and Ad Hoc DAC S/Cs). Re: Review FSER Chapter 7 and DACs.

4. November 3, 1993 (Safeguards and Security S/C). Re: Safeguards requirements for the ABWR.

5. November 16-17, 1993 (ABWR S/C). Re: Review FSER Chapters 3, 4, 6, 9, 15, and 17.

6. January 25-26, 1994 (ABWR S/C). Re: All remaining FSER Chapters and related matters and ACRS report.

7. February 1994. ACRS Full Committee for draft ACRS report.

DOCUMENTS

The review document for this Subcommittee meeting was the Advanced Boiling Water Reactor Standard Safety Analysis Report up to Amendment 31 (August 1993).

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NOTE: Additional meeting details can be obtained from a transcript of this meeting available in the NRC Public Document Room, 2120 L Street, NW, Washington, DC 20006, (202) 634-3273, or can be purchased from Ann Riley and Associates, Ltd., 1612 K Street, NW, Suite 300, Washington, DC 20006, (202) 293-3950.

MEETING OF THE ACRS SUBCOMMITTEE
ON ADVANCED BOILING WATER REACTORS
SEPTEMBER 8, 1993
BETHESDA, MARYLAND

APPROXIMATE TIME

1. Introductory Remarks — 8:30 - 8:45 a.m.
   • C. Michelson

2. Proposed Technical Resolutions of USIs and GIs — 8:45 - 10:15 a.m.

   • J. Fox et. al., GE

     - Per 52.47 (a)(iv)
     - Issues of special interest include USI A-17, 36, 43, 45, 47 and GSIs 43, 87

   * * * * * * **BREAK** * * * * * * * — 10:15 - 10:30 a.m.

3. Specification of Interface Requirements — 10:30 - 11:30 a.m.

   • E. Ehlert et. al., GE

     - Per 52.47(a)(vii), (viii) and (ix)
     - Using the ultimate heat sink and other appropriate examples:

       a. Show how the requirements are sufficient to allow completion of final safety analysis and design specific PRA

       b. Illustrate how they can be verified through inspection, testing, or analyses

       c. Show conceptual designs which are sufficient to complete a final safety analysis and PRA which will permit assessment of adequacy of the interface requirement

4. Closure of Open FSER Items and Schedules for ACRS Final FSER Review — 11:30 - 12:15 p.m.

   • C. Sawyer et. al., GE

   * * * * * * **LUNCH** * * * * * * * — 12:15 - 1:15 p.m.

- TENTATIVE SCHEDULE - (Cont'd)

		APPROXIMATE TIME
5.	Closure of Items Covered in Previous Subcommittee Meetings • J. Fox et. al., GE	1:15 - 3:00 p.m.
	* * * * * * **BREAK** * * * * * * *	3:00 - 3:15 p.m.
6.	Additional Questions on Final SSAR	3:15 - 4:30 p.m.
7.	Subcommittee Discussion and Adjournment	4:30 - 5:00 p.m.