Docket No. 52-001 May 13, 1994

Mr. Joseph Quirk
ABWR Certification Program Manager
GE Nuclear Energy
175 Curtner Avenue Mail Code - 782
San Jose, California 95125

Dear Mr. Quirk:

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

The purpose of this letter is to provide GE with additional staff comments on Amendment 34 to the ABWR standard safety analysis report (SSAR). Enclosure 1 contains a set of comments and SSAR pages with marked-up changes proposed by the staff. Enclosure 2 is a detailed discussion on two concerns resulting from the staff's review of Amendment 34. In order for the staff to complete the final safety evaluation report for the ABWR design, GE needs to provide an additional amendment to the ABWR SSAR that: (1) resolves the remaining items identified in my letter to GE dated May 3, 1994, and (2) addresses resolution of the primary containment pressure limit and the use of the containment spray as documented in Enclosure 2 of this letter.

For your information, Chester Poslusny is on rotational assignment, effective May 1, 1994, for six months, and Tom Boyce is acting lead Project Manager (PM) for the ABWR. Son Ninh and Dave Tang are the backup PMs. If you require any clarification or further guidance on these matters, please contact Son Ninh at (301) 504-1125 or Dave Tang at (301) 504-1147.

Sincerely,

(Original signed by Dennis M. Crutchfield for)

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

Enclosure:
As stated

cc w/enclosure:
See next page

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STAFF FEEDBACK ON ADVANCED BOILING WATER REACTOR (ABWR)
STANDARD SAFETY ANALYSIS REPORT (SSAR) AMENDMENT 34

SCSB Comments on SSAR

1. Table 3.9-8 (3.9-135): This table indicates that the drywell/wetwell vacuum breaker valves are tested every 6 months. This cannot be done. This was identified by SCSB (1.b) in Amendment 33, and GE stated in their summary submittal that they had corrected it.

2. Section 5.4.7.1.1.2 (5.4-31): This section incorrectly indicates the RCIC system instead of the RHR system.

3. Chapter 6: The table of contents needs to be revised.

4. Section 6.2.1.1.2.2 (6.2-5): Amendment 34 indicates that the suppression pool temperature will reach 76.7 C following a isolation event; whereas, Amendment 33 indicated 82.2 C. No basis was provided by GE for this change.

5. Section 6.2.1.1.3.3 (6.2-6): The following statement was added to Amendment 34: "Tolerances associated with fabrication and installation may result in the as-built size of the postulated break areas being 5% greater than the values presented in this chapter. These as-built variations would not invalidate the plants safety analysis presented in this chapter and Chapter 15." What is the basis for adding this statement?

6. Section 6.2.1.1.3.2 (6.2-6): The last two sentences are identical.

7. Section 6.2.1.1.8 (6.2-32): The last sentence repeats itself.

8. Section 6.2.1.3 (6.2-38): Reactor pressure value should be 7.31 MPaG, not 73.1 MPaG.

9. Section 6.2.5.2.6.2 (6.2-82): The median failure pressure is listed as 921.8 kPaG. Section 6.2.5.2.6.3 (6.2-82) indicates 931.7 kPaG. Section 6.2.5.2.6.6 (6.2-85) indicates 931.7 kPaG and 931.6 kPaG. These deviations were identified by SCSB (5e) in Amendment 33, and GE stated in their summary submittal that they had been corrected.

10. Section 6.2.5.6 (6.2-92): A reference to the sizing of the piping and rupture disk for overpressure protection is provided. This was identified by SCSB (6c) in Amendment 33, and GE stated in their summary submittal that this had been corrected.

11. Section 7.5.2.1(2)(o) (7.5-12): This section indicates that the drywell water level instrumentation cannot survive severe accident conditions. Table 19E.2-29 and Section 19E.2.1.2.3.3(12) indicates that they can survive.

Enclosure 1

12. Section 15.6.5.2.1 (15.6-8): This section references Table 6.2-8 for a sequence of events associated with barrier (containment) performance following an accident. However, Table 6.2-8 is a listing of primary containment penetrations and has nothing to do with barrier performance following an accident.

13. Section 18B (18B-15): The primary containment pressure limit is 0.56 MPaG, based on SRV operability as stated in Table 18B-1 PC/P-5. However, Section 19E.2.1.2.2.2(b) (19E.2-8) indicates that SRVs can operate at higher pressures. In particular, it states that the accumulators are charged to 1.27 MPa and that the actuators can open the SRVs with a minimum differential pressure of 0.482 MPa. This indicates that the SRVs are operable up to 0.79 MPa as opposed to 0.56 MPaG.

14. Figures 19E.2-28b (19E.2-290) need to be revised to show reactor vessel pressure following vessel melt-through.

15. Section 19ED.6 (19ED-25): The title "Satisfaction of Design Requirements" is incorrect. It should be "Related Experimental and Analytical Work."

Typos

1. Section 1.7: all references to Tables are incorrect.

2. Section 6.2.7.4 (6.2-102): "sell" should be "swell."

3. Table 6.2-7 (6.2-151): valve closure time should not have an "S."

4. Section 19E.2.1.2.3.1.1 (19E.2-12):

   - reference should be to Table 19.3-5.
   - reference should be to Figures 19E.2-26a through 19E.2-26e.

5. Section 19E.2.1.2.3.1.2 (19E.2-13):

   - references should be to Figures 19E.2-27a through 19E.2-27F.

6. Section 19E.2.1.2.3.1.3 (19E.2-14):

   - references should be to Figures 19E.2-28a through 19E.2-28f and 19E.2-29a through 19E.2-29f

7. Section 19E.2.1.2.3.3(2) (19E.2-19):

   - wording in first paragraph should refer to "process fluid" not "process steam."

8. Section 19E.2.1.2.3.3(10) (19E.2-22):

   - references should be to Figure 19E.2-28d and 19E.2-28a.

9. Section 19E.2.1.2.3.4 (19E.2-24):

   - references should be to Figures 19E.2-26a through 19E.2-26e, 19E.2-27a through 19E.2-27F, 19E.2-28a through 19E.2-28f, and 19E.2-29a through 19E.2-29f.

10. Section 19ED.4.5.1 (19ED-17):

   - discussions on sensitivity calculations should indicate +/- 20% and +/- 200K as indicated in Table 19ED-5.

SPLB Comments

1. GE needs to revise SSAR Table 3.9-8 to clarify that valve F175 supplied cooling water to the FPC heat exchanger, not the RHR heat exchanger.

2. GE needs to revise SSAR Figure 9.2-1, sheet 2 of 9, to identify this valve as F195A instead of F175A.

Additional Feedback on Amendment 34 Figures and Drawings Provided to ACRS in Marked-Up Versions

1. Figure 9.4-4 sheet 1 of 3 marked up changes were not included in Amendment 34.

2. Figure 9.4-4 (sheet 3 of 3) changes were not included in Amendment 34.

3. Figure 1.2-18 "ramp up to 4800mm" was deleted in Amendment 34.

4. Figure 1.2-22 deletions in upper drawing were not included in Amendment 34.

5. Figure 5.4-13 (sheet 1 of 2) valve designation for F019 was not deleted.

6. Figure 9.2-1 (sheet 5 of 9) changes were not included in Amendment 34.

The ABWR design satisfies this item.

1A.3 COL License Information

1A.3.1 Emergency Procedures and Emergency Procedures Training Program

Emergency procedures, developed from the emergency procedures guidelines, shall be provided and implemented prior to fuel loading (Subsection 1A.2.1).

1A.3.2 Review and Modify Procedures for Removing Safety-Related Systems from Service

Procedures shall be reviewed and modified (as required) for removing safety-related systems from service (and restoring to service) to assure operability status is known (Subsections 1A.2.18 and 1A.2.19).

1A.3.3 In-Plant Radiation Monitoring

Equipment and training procedures shall be provided for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during the accident (Subsection 1A.2.35).

1A.3.4 Reporting Failures of Reactor System Relief Valves

Failures of reactor system relief valves shall be reported in the annual report to the NRC (Subsection 1A.2.21.1).

1A.3.5 Report on ECCS Outages

Starting from the date of commercial operations, an annual report should be submitted which includes instance of ECCS unavailability because of component failure, maintenance outage (both forced or planned), or testing, the following information shall be collected:

(1) Outage date

(2) Duration of outage

(3) Cause of outage

(4) Emergency core cooling system or component involved

(5) Corrective action taken

The above information shall be assembled into a report, which will also include a discussion of any changes, proposed or implemented, deemed appropriate, to improve the availability of the emergency core cooling equipment (Subsection 1A.2.2.5).

1A.2.25

to Assure Safe Shutdown

Normal and Accident Equipment Environmental Conditions are cited in Appendix 3I. The Appendix 3I tables contained in this appendix identify environmental conditions at various plant building and equipment locations for a wide spectrum of plant design bases conditions. Pipe breaks both inside and outside primary containment are the dominant contributors to the abnormal plant conditions cited in the tables. Plant impact aspects from design basis fire or flood, or harsh environmental conditions have less of an impact. Due to divisional separation requirements enough equipment is isolated from the single plant fire and flood sources. That is, the event only affects a limited amount of equipment and plant area. Beyond design basis events plant effects like ATWS and SBO are enveloped by the above cited tables since the environmental conditions and effects of these events are less pronounced or momentary. Specific design requirements for these event demand inherent coping capabilities. A number of engineered safety features also mitigate the initial hostile conditions. Ultimately, other equipment is available to restore normal conditions, (e.g. CTG operation and restoration of HVAC). The subject Appendix 3I tables apply to safety-related equipment and their environmental qualifications. However, other equipment unaffected by these environmental conditions may also provide mitigation service (e.g., Turbine Building Feedwater).

3.13.7.2 Reactor Building Housed Equipment

The Reactor Building houses environmentally sensitive equipment in isolated and protected clean zones. These are areas which are not subject to design basis accident pipe break (inside or outside containment) effects. These clean rooms, areas or zones have their own independent and redundant component environmental control HVAC systems. The clean zones house a number of safety-related systems or related components (e.g. emergency electrical equipment rooms, the remote shutdown panel rooms, diesel generator rooms, etc.). The clean zones for redundant safety equipment are in themselves separated by divisional requirements related to fire, flood, and break aspects. Environmentally sensitive I&C equipment is housed in the Emergency Electrical Equipment (EEE) rooms. Not all equipment in clean zones are environmentally sensitive, In fact, only a small portion of the equipment are environmentally sensitive to changes in normal environmental conditions.

Safety-related RMUs and other MUX equipment are housed in EEE rooms. Severe plant event effects do not effect their safety functions. They are inherently unaffected by their own heat sources. They are also capable of prolonged loss of HVAC services due to their environmental locations and their low self heatup characteristics. Since there are three I&C divisions, environmental effects in one will not negate any demanded safety functions from the other locations.

3.13.7.3 Secondary Containment Housed Equipment

The Secondary Containment houses both safety-related and non-safety-related equipment. Little environmentally sensitive equipment is located inside the Secondary Containment. Although all equipment is ultimately affected by beyond Normal operation condition, the threshold EQ for most equipment is high and maximum event effect results are low to it. A limited number of potential pipe breaks inside the Secondary Containment require that housed safe shutdown equipment be designed and qualified for significantly elevated (above normal) environmental conditions. Even though these conditions are only momentary (a few seconds to a minute), equipment is qualified for them. The equipment is generally capable of operating for longer times at abnormal effect conditions than required by the design basis event effects.

No safety-related environmentally sensitive I&C equipment resides inside Secondary Containment (e.g. RMUs). Some non-safety related operational MUX equipment (e.g. RMUs) are housed in the Secondary Containment. Their failure or maloperation due to abnormal secondary containment conditions will not negate safety-related equipment abnormal event functions. The safety-related equipment in the RB/EEE rooms and the qualified safe shutdown equipment in the secondary containment will accomplish their safety function regardless of any non-safety system failures due to environmental conditions.

3.13.7.4 Divisional Separation Zones Housed Equipment

For most plant events and their effects the divisional separation zones generally afford another level of environmental protection and control. Each division has its own emergency HVAC system. For fire, flood and breaks inside containment, the divisional separation barriers assure complete independence, electrical, physical, environmental, etc. A small number of outside containment breaks limit the barriers effectiveness in regards to environmental effects. The equipment qualification requirements are designed to take these low probability events into account.

Less pronounced abnormal environmental conditions (e.g. divisional pipe leaks, fires, floods, HVAC loss, etc.) are readily isolated to the affected divisional zone and not allowed to propagate to the other divisional zones. Even postulated beyond design basis long term environmental effects (total loss of HVAC, extended SBOs, unisolated breaks, etc.) are accommodated. They are accommodated in the short term by the current conservative equipment environmental qualifications and alternative heat removal capabilities and in the long term by power recoveries, valve closures and break isolations, HVAC restoration and alternate heat removal systems.

3.13.7.5 Control Building Housed Equipment

The same protection afforded the above equipment is provided in the Control Building. Control Building environmental effects are induced and self-correcting. (e.g., Building ventilation exhaust radiation monitors during de-inerting, SGTS may be placed into service.

If purging (i.e., de-inerting) through the HVAC will [or does] result in a trip from the ventilation exhaust radiation monitors, then de-inerting will be [re-]initiated at a reduced rate through the SGTS. Use of SGTS during de-inerting is expected to be infrequent. butterfly

primary containment

The design basis condition for the relevant dose analyses assumes that the large ventilation valves are closed, because the probability of a LOCA occurring at the same time the ventilation valves are open is very small. The large ventilation valves are, in fact, closed throughout normal plant operation except during inerting and de-inerting. The LOCA dose analyses do not assume any release from open containment isolation valves, either through the SGTS or through the normal ventilation system.

Isolation valves are designed to accept such conditions & reliably close

A realistic assessment of plant capability in support of the exclusion indicates that the ventilation valves, if open, would be isolated before significant fission products are transported to the containment atmosphere. "Significant" means fission products above that normally present in the primary system. A period much longer than the closing time of the ventilation valves would be required to generate conditions leading to the release of TID 14844-like source terms. Therefore, should a LOCA occur when the ventilation valves are open (valves expected to be open only during inerting or de-inerting), little fission product release to the environment would actually occur. Therefore, the plant design and analysis in this regard is conservative and bounds releases actually expected in the event of a LOCA. Risk analysis was performed to identify increase in public risk for 15% power, open containment. It was only a 2% increase in the overall risk.

Also, the valves are designed to be closed by blowdown assistance

Am 34

6.5.1.4 Tests and Inspection

The SGTS and its components are periodically tested during construction and operation. These tests fall in three categories:

(1) Environmental qualification tests

(2) Acceptance tests as defined in ASME N509 and N510

(3) Periodic surveillance tests

The above tests are performed in accordance with the objectives of Regulatory Guide 1.52 and its references. Acceptance tests (including pre-operational tests) and periodic surveillance tests are defined and extensively described in ASME N509 and ASME N510. Testing requirements in ASME N509 are generally located in Section 5, "Components." ASME N510 provides details of each component functional test. These tests are summarized in Table 9-1 of ASME N509 and Table 1 of ASME N510. Specific surveillance testing requirements for SGTS are provided in Technical Specification 3.6.4.3 (Chapter 16). Environmental qualification testing is discussed in during system startup. It is also installed, as required, to preserve the integrity of the drainage systems. Floor drains in areas not restricted because of potential radioactivity are provided with caulked or threaded connections.

INSERT A

(5) **Cleanouts**—In collection system piping from areas of potential radioactivity, cleanouts are provided, when practicable, at the base of each vertical riser where the change of direction in horizontal runs is 90°, at offsets where the aggregate change is 135° or greater, and at maximum intervals of ~~50 feet~~. 15.2 m. Equipment hubs and floor drains are also used as cleanout points. Cleanouts are welded directly to the piping and located with their access covers flush with the finished floor or wall.

INSERT B

9.3.8.2.4 Safety Evaluation

The Drain Transfer System is not safety-related. ~~Sumps designated as containing radioactive wastes are equipped with charcoal filters in the vents.~~ In the event of a LOCA signal, all drywell sumps are automatically isolated to preclude the uncontrolled release of primary coolant outside the PCV.

This last Sen not deleted. Am.

9.3.8.2.5 Tests and Inspections

Drywell and Reactor Building floor and equipment drain sumps are provided with the following instruments and controls:

(1) High and low level switches are provided on each sump pump to start and stop the sump pump automatically. A separate high-high level switch set at a higher level starts the second pump and simultaneously actuates an alarm in the main control room.

(2) Leak detection is effected by monitoring the frequency and duration of pump runs.

9.3.9 Hydrogen Water Chemistry System

9.3.9.1 Design Bases

9.3.9.1.1 Safety Design Basis

The Hydrogen Water Chemistry (HWC) System is non-nuclear, non-safety-related and is required to be safe and reliable, consistent with the requirement of using hydrogen gas. The hydrogen piping in the Turbine Building shall be designed in accordance with the guidance Regulatory Guide 1.29 "Seismic Design Classifications", Section C.2 to comply with modified BTP CMEB 9.5-1, Part C.5.d(5).

9.3.8.1.1 Safety Design Bases

(1) The Drain Transfer System (DTS) drains equipment and floor areas where required for structural loading reasons and to protect systems required for a safe shutdown.

(2) All potentially radioactive drains are piped directly to the radwaste system and shall not affect safety-related equipment operation.

(3) Containment and drywell penetrations shall be designed and fabricated in accordance with the ASME Code, Section III, Class 2. These valves automatically close ~~open~~ when they receiving a LOCA signal. Secondary Containment penetrations shall be in accordance with the ASME Code, Section III, Class 3. Divisional Separation Zones piping and radwaste Tunnel connections will have check valves to preclude reverse flow + local isolation at the individual sumps.

NOT added in AM34

(4) Effluent from the radioactive drains shall be treated and monitored prior to discharge to assure that there are no unacceptable discharges.

(5) The radioactive drain transfer collection piping shall be provided with the following features:

(a) These piping systems shall be non-nuclear safety class and quality Group D with the exception of the containment penetrations and piping within the drywell, which shall be Seismic Category I and quality Group B. Additional exceptions are the backflow check valves in the ECCS equipment room sumps, which shall be Seismic Category I and quality Group C.

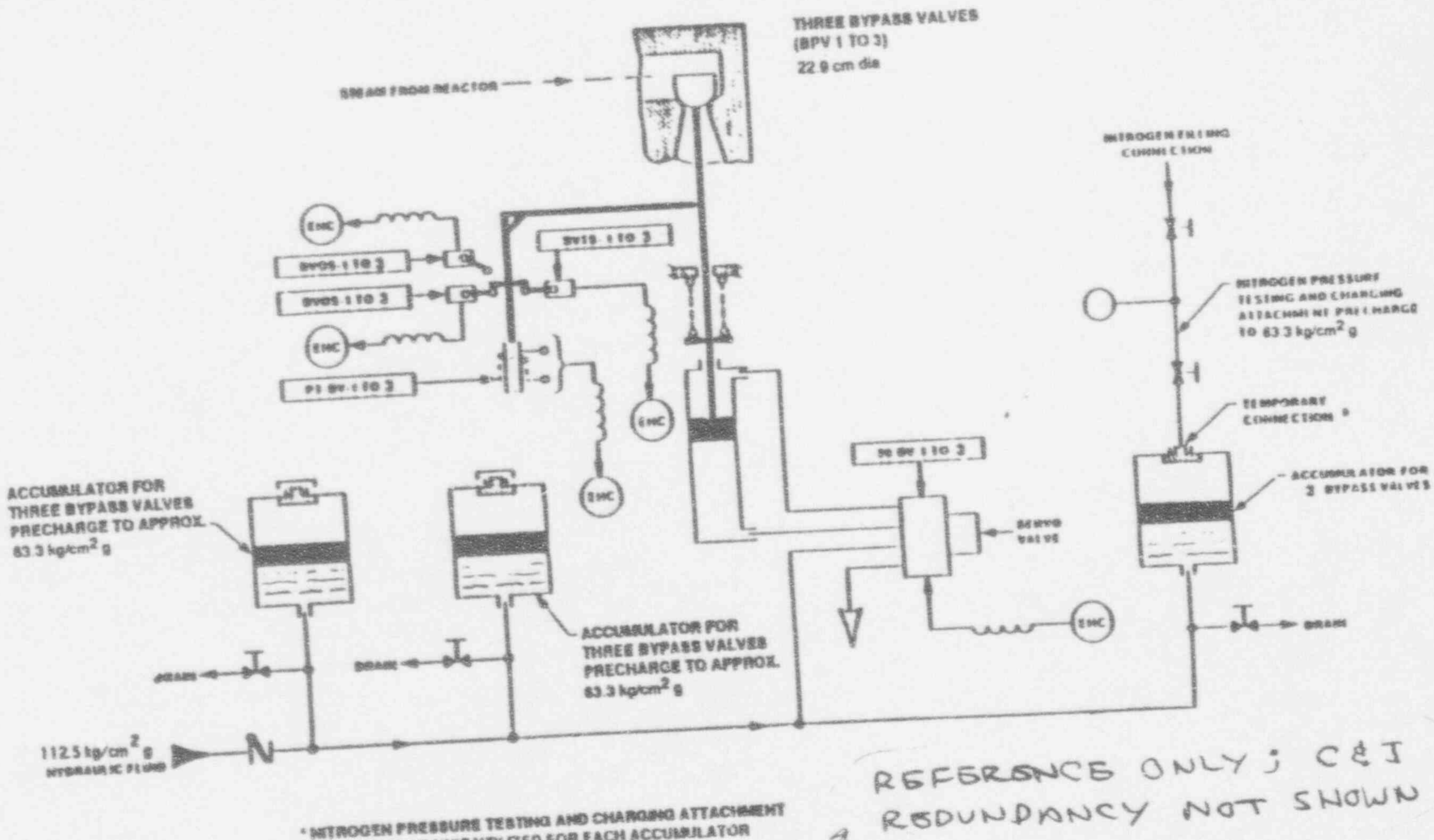
(b) The floor drain piping system in each divisional area of the ECCS pump rooms and the Control Building shall be arranged with a separate piping system for each quadrant or zone. The piping shall be arranged so that flooding or backflow in one quadrant cannot adversely affect the other quadrants.

(c) The COL applicant will provide equipment and floor drain piping Table of interconnection P&IDs for the part of the radioactive drain transfer system. From the Radwaste Tunnel to the Radwaste Building Tankage See Subsection 9.3.12.4 for COL license information requirements.

(d) There shall be no interconnection between any portion of the radioactive drain transfer system and any non-radioactive waste system which will permit transfer of radioactive material to the non-radioactive system. Effluent from non-radioactive systems shall be monitored prior to discharge to assure that there are no unacceptable discharges.

(e) Any valves that are relied upon to prevent backflow shall be inspectable and testable and designed to withstand SSE.

f) Reactor Building (Primary + Secondary & Divisional Separation Zone HCW Sumps will be treated prior to transfer into the Radwaste Tunnel. LCW Sumps will be likewise treated.



REFERENCE ONLY; C&I REDUNDANCY NOT SHOWN

NOT in Am 34

Figure 10.4-9 Bypass Valve Control, Electro-Hydraulic Control Unit

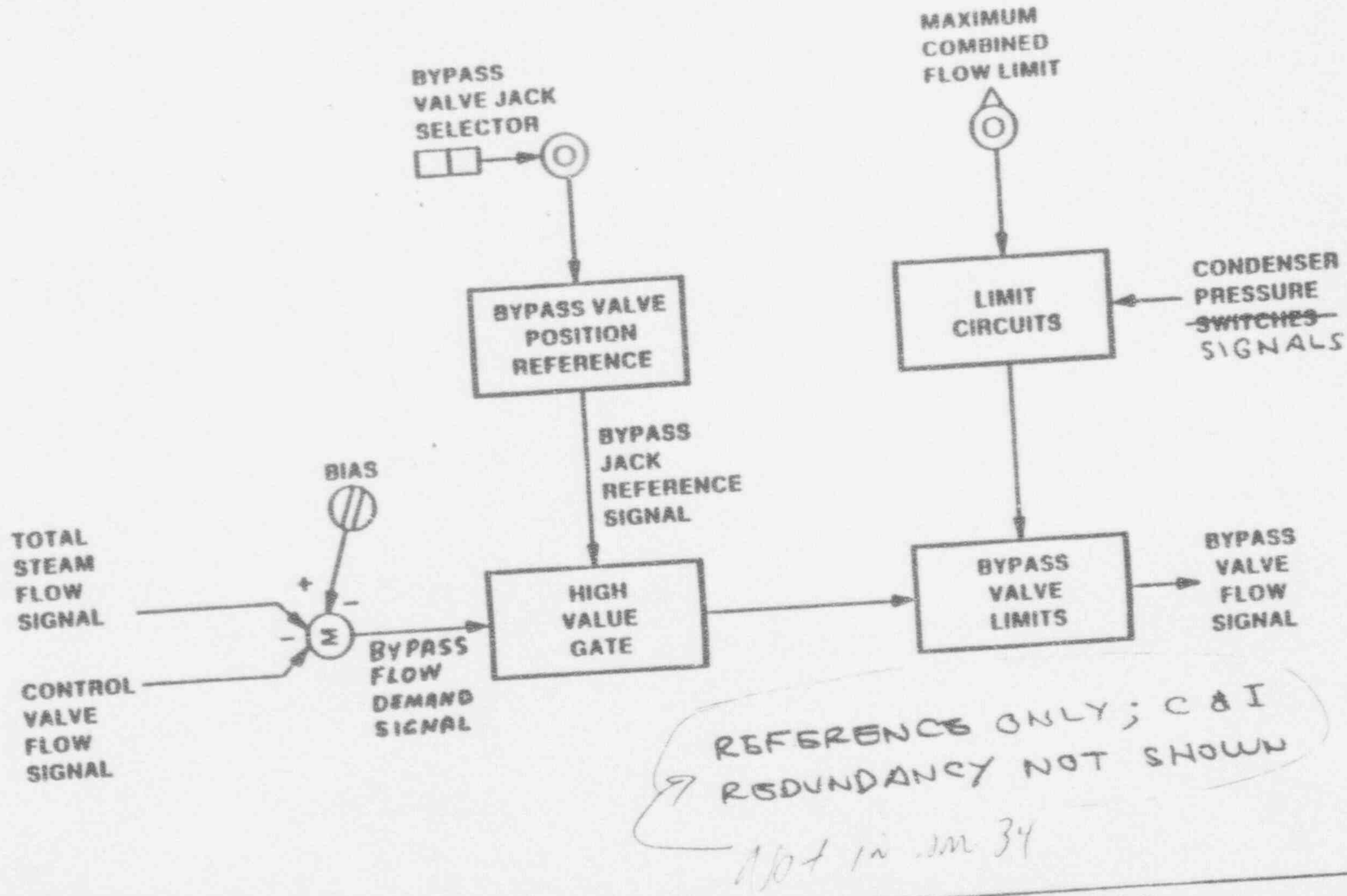


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Figure 10.4-10 Signal Flow Chart for Turbine Bypass Control Unit

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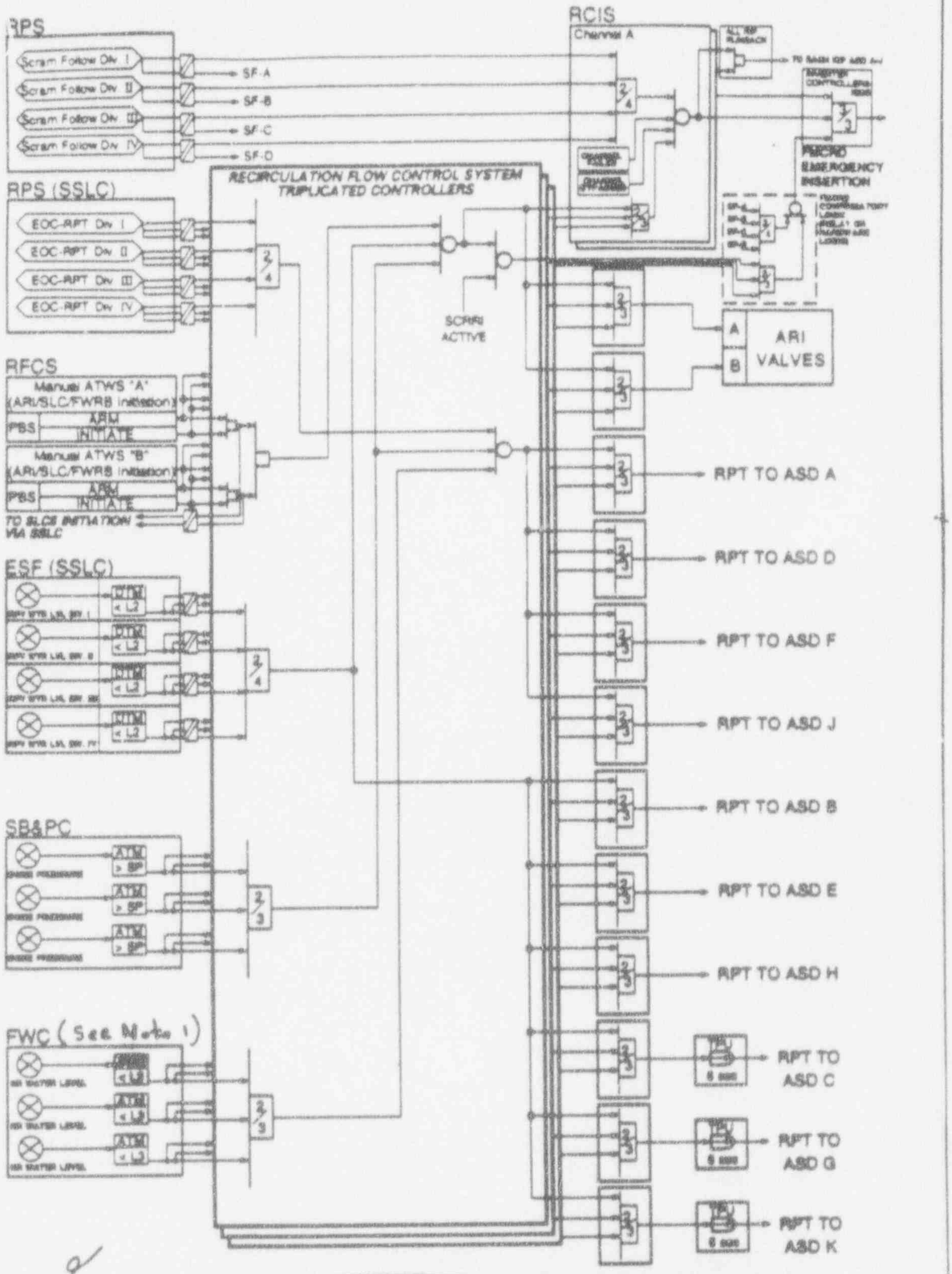


FIGURE 15E-1 ATWS MITIGATION LOGIC (ARI, FMCRD RUN-IN, RPT, RECIRCULATION RUNBACK)

Notes:
1. 4-pump RPT on L3 is not an ATWS mitigation event

Figure 15E-1 ATWS Mitigation Logic

a

Note: Not Added to Figure in Am 34

16.1 COL License Information

This section outlines the information required to be provided by the COL applicant to complete its plant specific Technical Specifications.

16.1.1 COL Information Required for Plant Specific Technical Specifications

In cases where the detailed design, equipment selection, or other efforts are required to establish the information to be specified in Technical Specifications, "[ ]" has been indicated. The COL applicant will evaluate their applicability and provide the required information to complete its plant specific Technical Specifications.

As part of the Technical Specification Improvement Program undertaken by the NRC and the industry, portions of Section 5.0, Administrative Controls, of NUREGs 1433 and 1434, could be relocated to licensee-controlled document. This improvement has been incorporated into the ABWR Technical Specifications. The COL applicant will have to ensure that the portions of Section 5.0 relocated to licensee-controlled documents are controlled in accordance with an administrative control system acceptable to the NRC.

Need to include in SSAR Table 1.9-1 and cross-reference in 16.1.1 above.

19.9.20 Actions to Assure Reliability of the Supporting RCW and Service Water Systems

To assure the reliability of the RCW and Service Water Systems, the COL applicant will take the following action. At least each month, the standby pumps and heat exchangers are started and the previously running RCW and service water equipment is placed in a standby mode.

19.9.21 Housing of ACIWA Equipment

If AC-independent water addition (ACIWA) equipment is housed in a separate building, that building must be capable of withstanding site specific seismic events, flooding, and other site-specific external events such as high winds (e.g., hurricanes). The capability of the building housing the ACIWA equipment must be included in the plant-specific PRA.

19.9.22 Procedures to Assure SRV Operability During Station Blackout

To assure the operability of the SRVs during station blackout, the COL applicant will develop procedures for the use of the stored nitrogen bottles as discussed in Subsection 19E.2.1.2.2.2 (b).

19.9.23 Procedures for Ensuring Integrity of Freeze Seals

boundary?

The COL applicant will provide administrative procedures to ensure the integrity of the ~~temporary~~ bounding when freeze seals are used. Mitigative measures will be identified in advance, and appropriate back-up systems will be made available to minimize the effects of a loss of coolant inventory (See Subsection 19Q.8).

19.9.24 Procedures for Controlling Combustibles During Shutdown

The COL applicant shall provide administrative procedures for controlling the combustibles and ignition sources during shutdown operations. (See Subsection 19Q.6 under "Fires During Maintenance").

19.9.25 Outage Planning and Control

The COL applicant shall provide an outage planning and control program to ensure that the safety principle is clearly defined and documented (See Subsection 19Q.10).

19.9.26 Reactor Service Water Systems Definition

Service water systems modeled in the ABWR PRA are described and fault trees presented in Subsection 19D.6.4.2. These include the Reactor Building Cooling Water (RCW) System, Reactor Service Water (RSW) System, and the Ultimate Heat Sink (UHS). Those portions of the RSW System that are outside of the Control Building and the entire UHS are not in the scope of the ABWR Standard Plant. The COL applicant

New Mark-up

Safety Issues Index (Continued)

Title	NRC Priority	SSAR Subsection
I.C.8 Pilot-Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	Resolved	COL App.
I.D.1 Control Room Design Reviews	Resolved	1A.2.2
I.D.2 Plant Safety Parameter Display Console	Resolved	1A.2.3
I.D.3 Safety System Status Monitoring	Medium	19A.2.17
I.D.5(2) Plant Status and Post-Accident Monitoring	Resolved	19B.2.65
I.D.5(3) On-Line Reactor Surveillance System	Near Res.	19B.2.66
I.F.2(2) Include QA Personnel in Review and Approval of Plant Procedures	Resolved	19A.2.43
I.F.2(3) Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Resolved	19A.2.43
I.F.2(6) Increase the Size of Licensees' QA Staff	Resolved	19A.2.43
I.F.2(9) Clarify Organizational Reporting Levels for the QA Organization	Resolved	19A.2.43
I.G.1 Training Requirements	Resolved	1A.2.4
I.G.2 Scope of Test Program	Resolved	19B.2.67
II.B.1 Reactor Coolant System Vents	Resolved	1A.2.5
II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	Resolved	COL App. 1A.2.6
II.B.3 Post-Accident Sampling	Resolved	1A.2.7
II.B.4 Training for Mitigating Core Damage	Resolved	COL App.
II.B.8 Rulemaking Proceeding on Degraded Core Accidents	Resolved	19A.2.1
II.D.1 Testing Requirements	Resolved	1A.2.9
II.D.3 Relief and Safety Valve Position Indication	Resolved	1A.2.10
II.E.1.3 Update Standard Review Plan and Develop Regulatory Guide	Resolved	COL App.
II.E.4.1 Dedicated Penetrations	Resolved	1A.2.13
II.E.4.2 Isolation Dependability	Resolved	1A.2.14
II.E.4.4 Purging	Resolved	19A.2.27
II.E.5.1 Test Adequacy Study	Resolved	19B.2.68 COL App.
II.F.1 Additional Accident Monitoring Instrumentation	Resolved	1A.2.15
II.F.2 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	Resolved	1A.2.16

19B.2.25-2 NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition*, U.S. NRC.

19B.2.25-3 *ASME Boiler and Pressure Vessel Code*, Sections III (Nuclear) and XI, American Society of Mechanical Engineers.

19B.2.26 B-66: Control Room Infiltration Measurements

Issue

Issue B-66 in NUREG-0933 (Reference 19B.2.26-1) addresses maintenance of the control room in a safe habitable condition under accident conditions by providing adequate protection for the plant operators against airborne radiation and toxic gases.

The rate of air infiltration into the control room is a significant factor in maintaining habitability, and the NRC measured air exchange rates in selected operating reactor plant control rooms to improve the data base for evaluating its effects.

No new design requirements were established by the NRC as a result of this and other work related to control room habitability in an accident. However, more specific review procedures were incorporated in SRP Sections 6.4.1, 9.4.1 and 15.6.5.5 (Reference 19B.2.26-2), including the habitability review provisions of TMI Action Plan Item III.D.3.4 (Reference 19B.2.26-1) regarding analyses of toxic gas concentrations and operator exposures from airborne radioactive material and direct radiation, to ensure more effective implementation of existing requirements.

Acceptance Criteria

The acceptance criteria for the resolution of Issue B-66 is that the control room ventilation and air-conditioning systems be designed to maintain the room's environment within acceptable limits for the operation, testing and maintenance of the unit controls and for uninterrupted safe occupancy during normal and accident conditions. Specifically, these systems shall be designed to meet the intent of the guidance given in SRP, Sections 6.4.1, 9.4.1 and 15.6.5.5.

6.5.1

Resolution

The ABWR main control area envelope is heated, cooled, ventilated and pressurized with respect to the atmosphere and adjacent areas are maintained at positive pressure with respect to the atmosphere by a system mixing recirculated air with filtered outdoor air. It is designed to ensure that the operators can remain in the main control area envelope and take actions to operate the plant safely under normal conditions and maintain it in a safe condition during and following an accident. There are two air intakes on the top floor side walls of the control building, one on each end. Redundant radiation monitoring sensors in each air intake warn operators of airborne contamination, and cause the CRHA HVAC system to switch automatically to an emergency system employing HEPA and charcoal filters for cleanup.

This control room habitability area heating, ventilating and air-conditioning (CRHA HVAC) system is designed:

- With redundancy to ensure operation in an emergency with a single, active failure;
- For radiation exposure limits not exceeding the guidelines of 10 CFR 50, Appendix A, General Design Criterion 19 (Reference 19B.2.26-3), for any of the Chapter 15 DBAs;
- With provisions to detect and remove smoke and airborne radioactive material;
- To provide a controlled temperature and pressurized environment for continued operation of safety-related equipment under accident conditions;
- Protection from toxic chemical and chlorine releases.

In addition, the safety-related components of the CRHA HVAC system are operable during loss of offsite power conditions using divisional onsite power from the diesel generators and safety-related batteries. Provisions are also made for periodic tests of the emergency filtration unit fans and filters. The high-efficiency particulate air (HEPA) filters of the CRHA HVAC system will be tested periodically with dioctyl phthalate (DOP) smoke. The charcoal filters will be periodically tested with an acceptable gas for bypasses. The system ductwork and housings, which are of welded construction, will be periodically tested for unfiltered inleakage in accordance with ASME N510.

This ABWR CRHA HVAC and its design bases are described in Section 6.4, 6.5.1 and Subsection 9.4.1.

Since the control room is monitored, pressurized and filtered by the above described systems, and since the NRC requirements and the guidance for their design are met, the issue of air infiltration is resolved for the ABWR.

References

19B.2.26-1 NUREG-0933, *A Prioritization of Generic Safety Issues (with Supplements 1- 15)*, U.S. NRC, April 1993.

19B.2.26-2 NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition*, U.S.NRC.

19B.2.26-3 10 CFR 50 Appendix A, *General Design Criteria for Nuclear Power Plants*, Office of the Federal Register, National Archives Records Administration.

Therefore, this issue is resolved for the ABWR Standard Plant design.

References

19B.2.34-1 NUREG-0933, *A Prioritization of Generic Safety Issues (with Supplements 1- 15)*, U.S. NRC, April 1993.

19B.2.34-2 Regulatory Guide 1.151, *Instrument Sensing Lines*, U.S. NRC.

19B.2.34-3 NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants—LWR Edition*, U.S. NRC.

19B.2.34-4 10 CFR 50 Appendix A, *General Design Criteria for Nuclear Power Plants*, Code of Federal Regulations, Office of the Federal Register, National Archives and Records Administration.

19B.2.34-5 ISA-S67.02, *Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants*, Instrument Society of America.

19B.2.35 51: Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

Issue

Issue 51 in NUREG-0933 (Reference 19B.2.35-1), identifies the susceptibility of the Station Service Water System (SSWS) to fouling which leads to plant shutdowns and reduced power operation for repairs.

The SSWS cools the Component Cooling Water System (CCWS) through the Component Cooling Water Heat Exchangers and rejects the heat to the ultimate heat sink (UHS) during normal, transient, and accident conditions. The CCWS in turn provides cooling water to those safety-related components necessary to achieve a safe reactor shutdown, as well as to various non-safety reactor auxiliary components.

Acceptance Criteria

Elimination of the possible effects of fouling of the service water system and ultimate heat sinks is a design goal of the ABWR. The ~~Plant Designer~~ COL applicant is given specific requirements and guidance on achieving this goal, including instruction to consider designs and new requirements which further mitigate the fouling effects. Additionally, the ~~Plant Designer~~ COL applicant is directed to investigate the problem with ice as a flow blockage mechanism and to dispose of and/or dissolve such ice as required.

Resolution

A review of operating plant experience shows that the most prevalent problems with plant cooling water systems are due to the corrosion and fouling caused by poor quality service water. In spite of a variety of water treatment schemes and use of expensive material, the wide range of harsh chemistry, silt and biological content result in a need

Acceptance Criteria

The acceptance criteria for the resolution of Issue I.D.5(2) is that plant status and post-accident monitoring is in compliance with Regulatory Guide (RG) 1.97 (Reference 19B.2.65-2).

Resolution

The ABWR design of its information systems (important to safety) provide information for manual initiation and control of safety systems. These systems provide indication to the control room that plant safety functions are being accomplished and provide information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences and accidents. It is designed to perform as described in Subsection 7.5 and is in compliance with RG 1.97 (Reference 19B.2.65-2).

Therefore, this issue, I.D.5(2), is resolved for the ABWR.

References 198c

19B.2.65-1 NUREG-0660, *NRC Action Plan Developed as a Result of the TMI-2 Accident*, U.S. NRC, May ~~1990.~~

19B.2.65-2 Regulatory Guide 1.97, *Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs condition During and Following an Accident*, U.S. NRC.

19B.2.66 I.D.5(3) On-Line Reactor Surveillance System

Issue

NUREG-0933 (Reference 19B.2.66-1), Generic Safety Issue (GSI) Item I.D.5(3) addresses the TMI issue of an "On-Line Reactor Surveillance System". This issue specifically concerns detecting abnormal reactor core internal's noise associated with on-line reactor operation, e.g., detecting loose internal reactor parts.

Acceptance Criteria

The acceptance criteria for the resolution of GSI-I.D.5(3) is that, based on the on-going generic BWR programs, it is concluded that the technical resolution of this issue has been identified (see Reference 19B.2.66-1).

Resolution

The primary cause of core vibration is high and turbulent reactor water recirculation flow. To detect such vibration, the ABWR design incorporates a reactor vessel loose parts monitoring system (LPMS), as described in Subsection 4.4.3 that complies with NRC's Regulatory Guide 1.133 (Reference 19B.2.66-2) requirements. In addition, with the redesign for the ABWR reactor core internals, i.e., core fuel supports, fuel boxes and instrument channel's etc., problem reoccurrence has essentially been eliminated. The LPMS and other ABWR instrumentation systems will continue to monitor various reactor operational parameters, e.g., reactor core vibration, neutron flux patterns and leakage). This is attributable to equipment and power recovery prior to containment failure and to "passive mitigation," i.e., flooding of the molten core from the suppression pool water when passive flooder system actuates.

Considering only those accident sequences in which core melt starts (i.e., exclude certain class II events where core melt was not initiated), then the core melt arrest constitutes approximately 86% of all such sequences. The frequency of core damage with significant fission product release, which includes all categories except NCL and OK, is 2.2E-8 per reactor-year.

The containment design incorporates a containment overpressure protection system which is designed to ensure that any sequence which is designed to ensure that any sequence which is not arrested in the containment will have low consequences. This system consists of a line originating in the wetwell which exhausts to the plant stack. If the containment pressure rises to a level where containment integrity could be challenged, a rupture disk opens relieving the containment pressure. If there is no suppression pool bypass, the containment does not reach the rupture disk setpoint for about 24 hours. This ensures a late release with low magnitude. The frequency of these

15

events is 2.1E-8, or 13% of all core damage events. The frequency of all other events is only 1.2E-9. Thus, the upper bound for releases with the potential to be early or have high magnitude is 0.8%.

19D.5.12.4 Probability of Containment Structural Failure Due to Loss of Heat Removal

One of the goals of the ABWR design is to assure that highly reliable heat removal systems be provided to reduce the probability of containment failure by loss of heat removal.

The frequency of containment structural failure resulting from a loss of containment heat removal systems is evaluated to be 1.1E-9 per reactor-year. Core damage occurs in only 0.2% of these events. This low number demonstrates that the goal is met for the ABWR design. The ABWR features and other factors that contribute to this low value are:

THE TREE FIG 19D.5-10 SUPPORTS 0.1 %. IT IS UNCLEAR WHERE THIS 0.2% COMES FROM

(1) Three divisions of heat removal systems.

(2) Ability to re-establish the main condenser as a heat sink in certain accidents.

(3) Ability to remove heat using CUW heat exchanger.

(4) Long times before containment pressure reaches a value which could threaten containment integrity, which enables recovery of power and failed heat removal systems.

(5) Presence of the containment overpressure protection system.

checking the availability of the HPCF system. If RCIC is successful, availability of the HPCF system is not examined. Thus, with the event tree structure in the baseline PRA, it is not possible to identify what percentage of Class II accident frequencies consist of sequences in which RCIC is the only system operating.

For the sensitivity study, the accident event trees were revised to examine the availability of the HPCF system prior to checking the availability of the RCIC system. Thus, the sequences with HPCF failure and RCIC success represent the contribution of RCIC alone to Class II CDF. Figure 19D.5-31 shows the revised event tree for Reactor Shutdown as an example of the modification made to the event trees.

Such a revision to the event tree does not affect the overall logic for the PRA and has no effect on the total plant CDF. However, this model enables identification of the frequency of Class II events in which RCIC is the only system in operation for core cooling. The accident frequency is calculated by the CAFTA computer code using Boolean logic operation. The CDF Value shown in the new column is an estimated value obtained by multiplying the Class II accident frequency by 1.0E-04 to account for recovery feed back from the containment response analysis. No further reference is made in this study to the CDF values in this column.

1.1 In order to obtain a more accurate estimate of the core damage frequency resulting from Class II scenarios where RCIC is the only available means of core cooling, a modified containment event tree (CET) was constructed (see Figure 19D.5-32). The first event in Figure 19D.5-32 (II) represents the frequency of Class II events occurring as calculated from the modified Level I analysis. The total frequency is calculated by summing the accident frequencies of all Class II sequences which yields a value of 1.0E-06. The CET provides a more detailed analysis of the containment response, thus giving the CDF for Class II events.

The second event in Figure 19D.5-32, RCIC ONLY, represents the fraction of all Class II sequences with only RCIC available. The event probability used, 0.017, is a direct result of the revised accident event trees discussed above, indicating that approximately 1.7% of all Class II sequences have only RCIC available for core cooling. Following the second event, the top half of the Figure 19D.5-32 tree (i.e., core cooling is not supplied by RCIC alone) is quantified identically as the baseline Class II CET presented in Subsection 19D.5.11.4, Figure 19D.5-10. The structure of the top half requires success of one or more of the following systems: HPCF, LPCF, Condensate pumps and/or Feedwater (FW) for the 24 hour mission time of the accident. The bottom half (i.e., RCIC only) must be quantified and structured in a slightly different manner as some of the event probability values depend on what other systems are available for core cooling.

The third event in Figure 19D.5-32 (RCH) assesses the probability of recovery of the long term heat removal systems. The top portion of the event tree in Figure 19D.5-32

THIS ANALYSIS IS LESS OPTIMISTIC THAN BASELINE ANALYSIS, BUT IS NOT TRULY CONSERVATIVE.

19D.5.14.3 Results

As expected, the results show that there was no appreciable change in the overall plant CDF. The baseline CDF due to Class II sequences, determined in Subsection 19D.5.11, is 1.1E-12 per reactor year. This amounts to only 7.0E-04 percent of the 1.6E-07 total baseline CDF per reactor year. From the modified Level 1 PRA event trees in this study, it was determined that 1.7% of the Class II sequences are cases in which only RCIC is available for core cooling. This result is not unexpected because of the availability of many reactor water makeup systems (HPCF, etc.). The ~~conservative~~ Level 2 sensitivity study reveals ~~that for cases in which RCIC is the only available method for core cooling,~~ a Class II CDF of 1.2E-10 ~~results~~. While this is a substantial increase over the baseline Class II CDF of 1.10E-12, it is still a very small fraction of the reported overall CDF of 1.6E-07. The overall risk would remain essentially unchanged, as approximately 98% of the CDF resulting from Class II sequences is attributed to sequences vented through the suppression pool. Only 2E-12 events per year would result in unscrubbed releases. Therefore, these results are not propagated into the overall reporting of results.

PRESENT WORDING IS CONFUSING. SUGGEST WORDING SUCH AS:

"THE BULK OF THIS FREQUENCY IS FROM EVENTS IN WHICH RCIC IS THE ONLY AVAILABLE MEANS OF CORE COOLING."

19E.2.1.2.3.1 Definition of Survivability Profiles

For each of the three categories of events, a set of curves representing the bounding environmental conditions for that category were developed for use in evaluating the equipment and instrumentation survivability. These conditions were then compared to the equipment capabilities to provide a measure of confidence that the necessary equipment would survive the expected conditions. It is important to note that the ABWR containment is inerted for all of the events described below. Therefore, there is no containment challenge due to hydrogen burning or detonation.

The basis for each category of events is provided below along with a brief summary of the event progression.

19E.2.1.2.3.1.1 10CFR50.34(f) Category

This category corresponds to an event which could result in the conditions of 10CFR50.34(f)(2)(ix), which specifies that core cooling is degraded sufficiently to result in the generation of 100% oxidation of the active cladding. Core cooling is then recovered before the vessel fails. The PRA has confirmed the results of previous studies which show that the core damage frequency is dominated by accidents initiated from transients. Table 19-3-5 indicates that only 0.4% of all core damage events are initiated by LOCA. Therefore, a transient initiated event is specified for this evaluation.

Best estimate analyses do not result in oxidation of 100% of the active cladding. In order to simulate the hypothetical event, MAAP-ABWR was run using a multiplier to non-mechanistically generate oxidation of the active cladding. Additionally, ECCS was cycled on and off to produce the requisite amount of hydrogen for 100% metal-water reaction. The event progresses as follows:

- An isolation event occurs.
- All core injection is assumed to fail.
- Drywell and wetwell sprays are initiated 30 minutes after the initiation of the accident, water flow is directed through the RHR heat exchanger.
- The core begins to heat up and zirconium begins to oxidize.
- ECCS is recovered.
- Additional hydrogen is generated as the core is quenched.
- Vessel water level is recovered, terminating the event.

Curves representing the environmental conditions during this event are shown in Figures 19E.2-25a through 19E.2-25e. The vessel pressure remains within the range of

SHOULD BE "19E.2-26a THROUGH 19E.2-26e"

A STATEMENT REGARDING WHY THIS IS NOT A CONCERN FOR RHR SHOULD BE ADDED HERE [UNDER (2)].

temperature of 377 K. As discussed above, however, the piping is nominally capable of withstanding pressures up to 2.5 times the rated pressure. The high suppression pool water temperature does not pose a problem for RHR system components because they contain no organic material. In the shutdown cooling mode, the RHR loop isolates from the RPV at 0.9 MPa. In the low pressure core injection mode, the RHR loop isolates from the RPV at 3.0 MPa. In-board of the isolation valves all components are rated to a pressure of 8.6MPa and a temperature of 575 K. Because of these ratings, severe vessel conditions do not threaten RHR survivability. Since the reactor pressure will not increase after RHR activation, overpressurization will not occur.

(3) Firewater System

The firewater system may be called upon to inject water into the vessel for a severe accident with in-vessel recovery or for the 10CFR50.34(f) event or through the drywell sprays during a severe accident which progresses ex-vessel. The system is manually initiated. All flow in the system is from outside the containment. Thus, accumulation of radioactive material in the firewater pumping system will not occur. All components of the firewater system are outside of the containment and will not be significantly affected during a severe accident. Inside the containment, the firewater system utilizes RHR valves, piping and spray headers which were discussed in (2).

(4) Passive Flooder

The passive flooder may be needed to provide a water flow path from the suppression pool to the lower drywell after vessel failure. The flow path is opened as a direct result of high temperatures in the lower drywell which occur after debris relocation from the vessel. This system does not contain any active systems, instrumentation or controls. Additionally, the system components are not hindered from performing their functions due to high radiation levels which might exist in the lower drywell after debris relocation from the vessel. Therefore, the system is expected to operate under the required conditions.

(5) Containment Overpressure Protection System (COPS)

The COPS may be needed during a severe accident to relieve high containment pressure. No credit is taken for the COPS system for the 100% metal-water reactor event. The system contains piping, a rupture disk and two valves which are normally open and fail open. To relieve containment pressure, the rupture disk must burst. Activation will not be adversely affected by the radiation in the wetwell airspace during a severe accident. The sensitivity of rupture disk activation to wetwell temperature is discussed in Subsection 19E.2.8.1.2.

However, if the arriving decompression does not cause the pool pressure to fall below its saturation value, flashing would not occur, and the pool would respond as a compressed liquid.

The theoretical modeling used to determine pool response from operation of the COPS includes prediction of:

- The gas discharge rate
- The velocity and decompression disturbances originating where the COPS enters the airspace
- Expansion of the decompression into the airspace, and its attenuation with distance
- Decompression transmission from the airspace into the pool at the water surface
- The pool water dynamic and thermodynamic response

It was found that the originating decompression wave entering the containment airspace was 38.8 kPa, dropping below the initial 721 kPa air pressure. The decompression wave leaving the COPS pipe of 0.275 m (0.9 ft) radius would reach the pool surface a distance of 4 m (13.12 ft) away, attenuating from 38.8 kPa to 2.67 kPa. Since sound speed and density of water are much higher than corresponding values in air, a decompression wave entering the water is nearly twice that arriving in the air, or about 5.34 kPa. The decompression is not large enough to cause pool pressure to drop below its saturation pressure of 330 kPa at its initial temperature of 410 K, or 137°C (738 R or 278°F). The pool surface would move upward at only 0.0044 m/s (0.014 fps) for the transmitted decompression.

SEE COMMENTS ON PAGE 19E.2-102

19E.2.3.5.1.2 The Gas Discharge Rate

The COPS pipe has a radius R and area A. The open COPS rupture disk has a flow area a. Since the airspace pressure P_o is 721 kPa and discharge is into the atmosphere at 101 kPa, the initial air flow is expected to be choked in the valve throat at a choked mass flux of (Reference 19E.2-37)

$$G_{gc} = \left(\frac{2}{k+1}\right)^{(k+1)/2(k+1)} \sqrt{k g_0 P_0 \rho_{g0}} \qquad (19E.2\text{-}41a)$$

The quasi-steady mass flow rate through the pipe and valve is expressed as

$$m = G_{gc} a \qquad (19E.2\text{-}41b)$$

Assuming isentropic flow from the airspace to the throat, and expressing the airspace sound speed as:

$$C_{g0} = \sqrt{(kg_0 P_0)/\rho_{g0}} \qquad (19E.2\text{-}41c)$$

the discharging mass flow rate is obtained in the form,

$$\frac{m}{AC_{g0}\rho_{g0}} = \left(\frac{2}{k+1}\right)^{(k+1)/2(k+1)} \frac{a}{A} \qquad (19E.2\text{-}41d)$$

19E.2.3.5.1.3 Disturbance Entering the Airspace

It is assumed that the COPS valve opens instantly, causing an instantaneous quasi-steady flow in the attachment pipe. This assumption gives the maximum pipe velocity, which corresponds to a maximum initial decompression wave.

Acoustic theory can be applied if pressure disturbances do not create Mach numbers much greater than 0.2. An area ratio of a/A = 0.132 (diameter ratio of d/D = 0.364) with an airspace state described by

SEE COMMENTS ON PAGE 19E.2-102

P_0 = 721 kPa

T_0 = 410 K (278°F)

g_0 = 6.16 kg/m^3 (0.384 lbm/ft^3)

C_{g0} = 406 m/s (1332 fps)

yields a gas velocity in the pipe of 31 m/s (102 fps). The corresponding mach number is 31/406 = 0.076, which justifies treating the decompression as an acoustic wave.

It is further assumed that the discharge begins suddenly, imposing the pipe flow velocity of 31 m/s at its entrance. In order to employ spherical propagation of the acoustic wave, an imaginary hemisphere of pipe radius R = D/2 = 0.55/2 m = 0.275 m (0.902 ft) has twice the pipe flow area, reducing the entrance velocity on the hemisphere to 31/2 = 15.5 m/s (50.8 fps). The acoustic equation,

$$\delta P_0 = \frac{C_\rho \delta V}{g_0} \qquad (19E.2\text{-}41e)$$

can be employed to show that the corresponding decompression disturbance is P_0 = 38.8 kPa (5.6 psid).

where all terms were defined previously and the subscript D refers to the conditions in the drywell.

The ratio of the flow rates from the drywell to pool flashing is found by combining Equations 19E.2-50 and 19E.2-51:

$$\frac{W_D}{W_P} = \frac{V_D M_{a,D} h_{fg}}{RT_D m_P \frac{dh_f}{dP}} \qquad (19E.2\text{-}52)$$

Pool swell is of chief concern for cases in which the firewater addition system has been used to add water to the containment. The suppression pool mass for this case is about 7.0E6 kg. An upper bound estimate of the mass flow ratio assumes that the drywell contains nitrogen at relatively low temperature 373 K (100°C) and that the suppression pool is hot ~~410 K (137°C)~~. Under these conditions the flow rate ratio is 0.043. These conditions will not occur in the ABWR, since the drywell cannot be cool when the containment pressure is high. However, this value is useful to gain an understanding of the range of Equation 19E.2-52. The bounding calculation shows that less than 5% of the flow through the COPS is being drawn through the horizontal connecting vents. Therefore, the primary contributor to pool swell is flashing of the suppression pool.

19E.2.3.5.3.4 Application to ABWR

The scenarios used in the suppression pool level swell calculations are identical to the accident sequences described in the ABWR SSAR Subsection 19E.2.2.1, Loss of All Core Cooling With Vessel Failure at Low Pressure (LCLP), leading to the opening of the Containment Overpressure Protection System Rupture Disk (R). These results are typical of all initiating events leading to the opening of the rupture disk. The passive flooder actuation scenarios will lead to the highest pool water temperature; thus the passive flooder cases are limiting for the onset of flashing. The firewater addition scenarios will lead to a higher water level swell for given thermal hydraulic conditions because the initial water height is higher.

NOT TRUE

2 MAJOR EXCEPTIONS ARE: (1) POOL BYPASS SEQS, AND (2) CLASS 2 SEQS. POOL BYPASS IS NOT A CONCERN BECAUSE POOL WILL BE COOL. CLASS 2 WOULD BE MORE LIMITING BECAUSE CST HAS BEEN ADDED AND POOL IS AT TSAT ~330°F (165°C)

Figures 19E.2-2 a-j show the drywell and wetwell conditions during a passive flooder actuation scenario. This scenario occurs when the passive flooder (PF) opens to cover the corium. This scenario leads to the maximum suppression pool water temperature. Figure 19E.2-2j shows that the maximum suppression pool temperature is 410 K.

Figures 19E.2-3 a-g show the drywell and wetwell conditions during a firewater addition scenario. This scenario occurs when the firewater system (FS) is actuated four hours after the initiation of the event. This scenario leads to the maximum suppression pool water level. Figure 19E.2-3f shows that the maximum suppression pool level is 14.5 m.

THIS SECTION SHOULD BE MODIFIED TO ADDRESS THE POOL SWELL ASSOCIATED WITH CLASS 2 SEQUENCES

137°C NOT LIMITING — 165°C IS

Pool swell is maximized at high temperature (~~410 K, 137~~°C) and high water level (14.5 meters, corresponding to an elevation of 1.35 m). The geometry of the containment and the bounding conditions are shown in Figure 19E.2-25. It is presumed that the rupture disk has just opened. Since the pool swell elevation is more sensitive to flow from the drywell, the upper bound value for the mass flow ratio found above is used. For the limiting suppression pool conditions, the average void fraction due to pool flashing is about 4%. This results in a pool swell of 0.65 meters, corresponding to an elevation of 2.0 meters. Since the bottom of the COPS penetration is at an elevation of 4.25 meters, this mechanism alone will not lead to flooding of the COPS penetration.

If the pool level were to rise an additional 2.25 meters near the outer wall of the suppression pool due to flow from the drywell, the COPS penetration could be flooded. A void fraction of 13% due to through flow from the drywell is required for this additional pool swell. Applying Equations 19E.2-47, 19E.2-48 and the upper bound value from Equation 19E.2-52, one arrives at an radius of 0.84 meters for the region affected by flow from the drywell. This area would be located near to the horizontal connecting vents at the inner wall of the suppression pool. Since the distance between the inner and outer walls of the suppression pool is 7.5 meters, one may safely conclude that pool swell will not threaten the COPS under these conditions.

TRAC calculations have been performed regarding suppression pool swelling during depressurization. TRAC uses two-fluid modeling instead of the drift flux model.

The TRAC level swell model has been qualified against test data. The PSTF experimental blowdown facility was used to provide information on liquid flashing due to a depressurization, and the subsequent swell of the liquid level. When compared with the TRAC model of the PSTF test, it was found that "the two-phase level comparisons show close agreement (± 10%)" (Reference 19E.2-39) The TRAC PSTF qualification validates the TRAC suppression pool swelling results.

A TRAC study of a typical Mark II containment (Reference 19E.2-36) showed a maximum pool level swell height of 0.79 m above the initial pool level. When the Mark II suppression pool level swell is calculated with the drift flux model used for the ABWR calculations, a maximum pool level swell of 2.33 m is obtained. This is almost three times as high as the TRAC two-fluid modeling results. This result demonstrates that the ABWR pool swell calculation is conservative.

19E.2.3.5.4 Carryover Due to Entrainment

The entrainment of water droplets by the steam flow through the suppression pool is potentially a concern since the water could carry fission products through the COPS to the environment. A very simple estimate analysis based on the work by Kutateladze (Reference 19E.2-18) indicates the potential entrainment for a pool of water sparged from below. The threshold for the entrainment of a droplet is based on the velocity of the steam from the surface of the suppression pool:

$$U_{threshold} = 2.7\left[\sigma g\left(\frac{\rho_1 - \rho_g}{\rho_g^2}\right)\right]^{1/4} \tag{19E.2-53}$$

Assuming the properties of steam at the rupture disk setpoint, the threshold velocity is about 6 m/s. The superficial velocity from the surface of the suppression pool is 0.02 m/s, assuming all of the flow through the COPS was passed through the suppression pool. Thus, there is more than two orders of magnitude between the superficial velocity which would be observed under the conditions of interest and the threshold for entrainment. This indicates there will be no significant entrainment from the surface of the pool.

A more sophisticated analysis is possible using the work of Rozen, et. al. (Reference 19E.2-17) to estimate even very low amounts of entrainment. This method uses the superficial velocity of steam rising from the pool and the pressure of the system to determine the typical droplet size and the ratio of liquid mass to vapor mass which is entrained from the surface of the pool.

INSERT NEW NUMBER BASED ON ANALYSIS OF CLASS 2 SEQS

For cases in which the firewater system has been used to add water to the suppression pool, the distance between the bottom of the COPS penetration (elevation 4.25 meters) and the ~~pool surface~~ (elevation 1.35 meters) is 2.9 meters. Assuming the maximum pool swell of 0.~~65 meters~~, discussed above, the height between the COPS and the pool surface is 2.25 meters. The correlation selected to calculate carryover is conservative for cases in which the water pool is at least two meters below the COPS penetration.

Using this correlation, the ratio of liquid mass to vapor mass is about 4E-6. If one considers an energy balance on the suppression pool before and after the rupture disk opens, it can be determined that just over one tenth of the suppression pool flashes to steam during the blowdown. Thus, the fraction of suppression pool liquid which might be transported from the suppression pool as a liquid is 4E-7.

The fission products in the suppression pool will exist as a dissolved salt and as sediment on the bottom of the pool. Therefore, the fraction of the fission products which can be carried out the COPS by entrainment will be some fraction less than the ratio of the liquid entrained from the pool surface. However, a release fraction of 4E-7 will not lead to significant offsite dose.

19E.2.3.6 Behavior of Access Tunnels

If core debris is entrained out of the lower drywell and into the access tunnels, it is possible that the integrity of the tunnels could be compromised. This depends on several key factors, such as the amount of debris entrained into the tunnels, whether the debris remains in the tunnels, the heat transfer characteristics between the debris and the tunnel walls, and the strength and loading of the tunnel material.

19E.2.3.6.1 Potential for Debris to Enter Tunnel

Based on the configuration of the lower drywell and the equipment contained therein, it is highly unlikely that debris will be carried into the tunnels unless there is significant debris entrainment. Based on work at INEL, (Reference 19E.2-35) local failure of the lower head is expected. In fact, the drain plug located at center of the bottom head appears to be the dominant failure location. A localized failure should result in a concentrated discharge from the center of the lower head. Immediately below the reactor vessel are the CRD mechanisms. Splashing off of the CRDs is not judged to result in a significant amount of debris transport to the tunnels. Since the debris is likely to be discharged from the center of the CRD array, radial movement through a forest of vertical structures is not expected and transport of the debris outside of the CRD array is not judged to be likely. In fact, the CRDs will tend to columnate the flow, since they are long, vertically oriented and have little change in cross section along their length.

Approximately 6 meters below the bottom of the vessel is the equipment platform constructed from thin steel grating material. This grating is located at about the elevation of the tunnel bottom. No other structures exist at or above this elevation to divert the discharging debris into the tunnel. The grating surface area is small compared to the overall cross sectional area of the lower drywell and the thermal properties of the debris would result in immediate melting of the grate. Further, the center of the equipment platform, where the debris is likely to flow, does not have any grating to allow movement of the CRDs during refueling. Thus, the presence of the equipment platform is not expected to result in significant splashing of the debris into the tunnels.

Using Ishii's methodology, debris entrainment thresholds were only reached for high pressure melt ejection events with very large vessel failure areas (Subsection 19EA.3.6.2). Based on work done at INEL and contained in the expert elicitations in NUREG 1150, and consistent with the DCH analysis in Attachment 19EA, a probability of 0.1 is assigned to a large (> 2 m^2) vessel failure area. Combining this with the probability of a high pressure core melt with melt ejection, this scenario constitutes only ~~2%~~ 3% of all core damage events. Thus, the potential for debris entrainment and the transport of debris to the access tunnels is judged to be quite low for the ABWR.

19E.2.3.6.2 Bounding Calculation Assuming Debris Enters Tunnel

Bounding calculations are performed to address those very low probability scenarios in which debris is transported into the tunnels.

BASED ON (0.27) × (0.1)

3% IS ALSO CONSISTENT WITH VALUE GIVEN ON P. 19EA-25

$$\int_0^t q''_{C,0}\,dt = \int_0^{19} q''_{St,0}\,dt + \rho_{St} C_{P,St} \int_0^{\delta_{St}} (1200 - T_{St}(x))\,dx \qquad (19E.2\text{-}57)$$

The debris is assumed to be infinite during this time, and the steel is assumed to behave as an infinite slab during the first 19 seconds. It is also assumed that the contact temperature is constant, at 1087 K, during the entire time. Solution of this indicates that the tunnel shell will reach 1200 K at approximately 46 seconds. Thus, this rather crude analysis indicates that the tunnel may fail in the unlikely event that debris is entrained.

19E.2.3.6.3 Impact of Tunnel Failure

Failure of the tunnel wall would occur at the lowest point. This would result in a flow path from the lower drywell vapor space into the suppression pool. As indicated earlier, there will initially be at least 1 meter of water above the bottom of the tunnel. Thus, no fission product bypass of the pool would occur. Since the event being considered is a high pressure melt scenario with entrainment of debris, the operator must initiate the firewater addition system in drywell spray mode to prevent high temperature failure of the drywell. This action will indirectly result in additional water being added to the suppression pool as it spills from the upper drywell, through the connecting vent system to the wetwell. Thus, several meters of water would be present above the tunnel failure elevation to provide scrubbing of fission products.

19E.2.3.6.4 Conclusion

It is unlikely for core debris to be entrained or splashed into the access tunnels. 3% Approximately ~~2%~~ of all core damage sequences could lead to debris entering the tunnels.

However, in the event that it does, the tunnel steel will reach temperatures that may compromise its integrity. The heat transfer through the thin steel wall is so high that the water on the outside of the tunnel quickly goes into dryout, and the heat can no longer be removed at a rate sufficient to maintain the tunnel integrity.

Failure of the tunnel wall would occur at the lowest point and would result in a fission product release path into the suppression pool. However, since several meters of water will be present above the tunnel failure site, fission products would be scrubbed and no containment bypass would result.

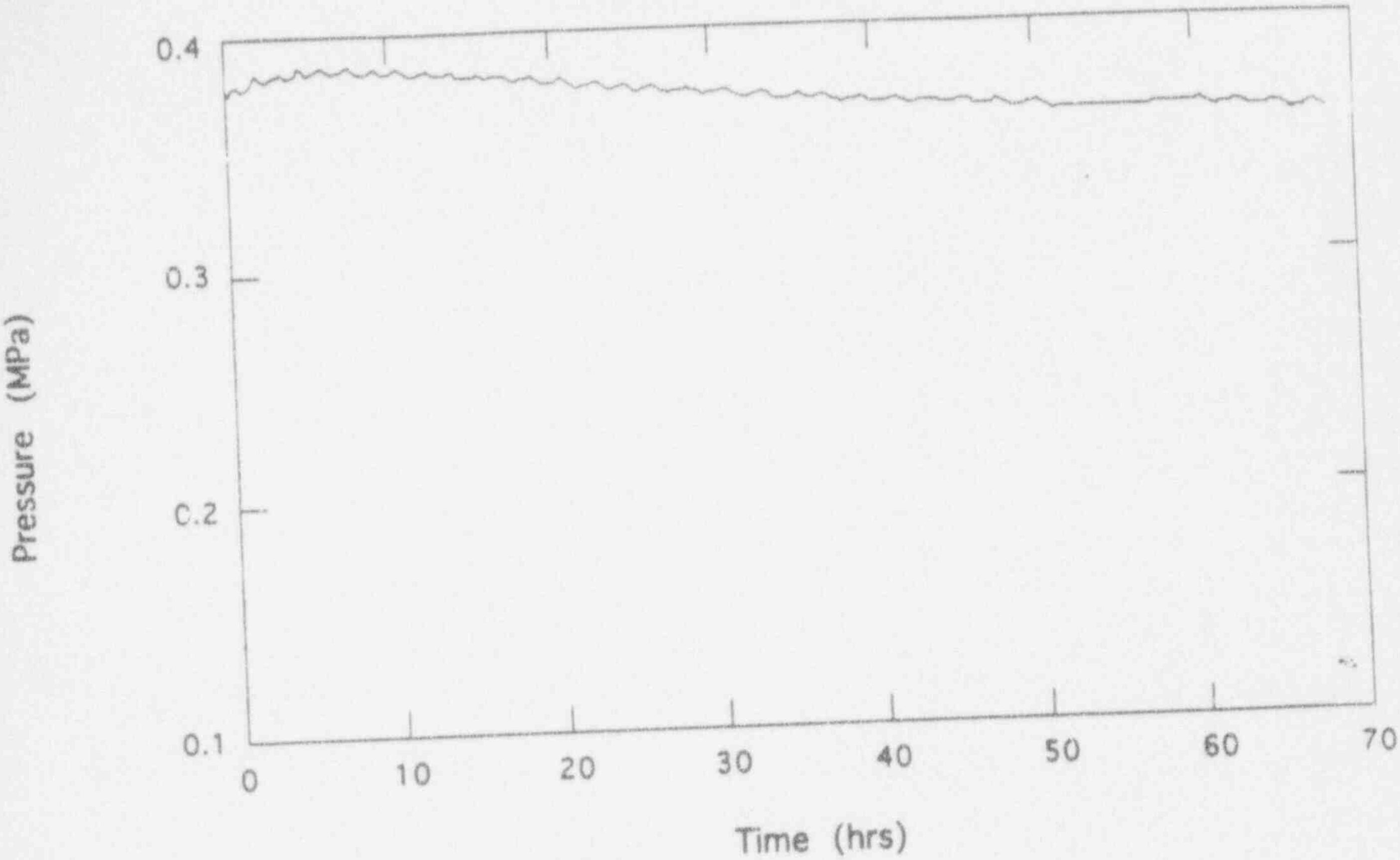
19E.2.4 Supplemental Accident Sequences

In order to quantify the PRA, sequences were analyzed using MAAP-ABWR to assess the effects of recovery. Additionally, some sequences with unusual characteristics, such as those having no containment structural failure, are considered in this subsection.

Table 19E.2-1 Potential Suppression Pool Bypass Lines

Description	Number of Lines	Pathway From	Pathway To	Pathway Size (mm) (1 in. = 25.4 mm)	Isolation Valves	Basis For Exclusion (See Notes)
Main Steam	4	RPV	ST	700	(AO, AO)	-
Main Steam Line Drain	1	RPV	ST	80	MO, MO	3
Feedwater	2	RPV	ST	550	CK, CK	-
Reactor Inst. Lines	37	RPV	RB	6	CK	-
CRD Insert	205	RPV	RB	1	CK, MA	1
HPCF Discharge	2	RPV	RB	200	CK, MO	-
HPCF Equalizing	2	RPV	RB	20	MO, MO	-
HPCF Suction	2	SP	RB	400	MO	2
Supp Pool Instrumentation	6	SP	RB	6	CK	2
SLC Injection	1	RPV	RB	40	CK, CK	-
RCIC Steam Supply	1	RPV	RB	150	(MO, MO)	-
RCIC Discharge	1	RPV	RB	150	CK, MO	5
RCIC Min. Flow	1	SP	RB	150	MO	2
RCIC Suction	1	SP	RB	200	MO	2
RCIC Turbine Exhaust	1	SP	RB	350	MO, CK	2
RCIC Turb. Exh Vac Bkr	1	SP	RB	40	CK, CK	2
RCIC Vac Pump Discharge	1	SP	RB	50	MO, CK	2
RHR LPFL Discharge	2	RPV	RB	250	CK, MO	-
RHR Equalizing Lines	2	RPV	RB	20	MO, MO	-
RHR Wetwell Spray	2	WW	RB	100	MO	2,4
RHR rywell Spray	2	DW	RB	200	MO, MO	4
RHR SDC Suction	3	RPV	RB	350	MO, MO	3
CUW Suction	1	RPV	RB	200	(MO, MO, MO)	-
CUW Return	1	RPV	RB	200	CV CV [redacted], [redacted] MO	5
CUW Head Spray Line	1	RPV	RB	150	CK, MO, MO	3
CUW Instrument Lines	4	RPV	RB	6	CK	-
Post Accident Sampling	4	RPV	RB	25	(MO, MO)	-
RIP Motor Purge	10	RPV	RB	<1	CK, CK	1
RIP Cooling Water	4	RPV	RB	200	MO, MO	1

not corrected



Where is this figure
19E.2-25?

Figure

"5" in text gDm34

determined to a first degree by comparing the Gibb's free energy of the oxides which make up the shield wall and the oxides present in core debris. Specifying alumina as the shield wall material satisfies this requirement.

(7) Seismic Adequacy

The seismic adequacy of the corium shields will be determined in the detailed design phase. Adequacy should be easily met because the shields are at the lowest point in the containment. Missile generation is not an issue because the shields are not near any vital equipment.

(8) Channel Height

The channel height is specified to be one centimeter.

Subsection 19ED.6 contains calculations which demonstrate satisfaction of requirements (1) through (4) for a chosen channel height of 1 cm.

19ED.4 ~~Analysis of Shield Freezing Ability~~ Channel Length

Heat transfer and phase change analyses are presented in this subsection to determine the ~~feasibility of a channeled shield to prevent~~ FDS channel length which prevents molten debris ingression into the ~~floor~~ drain sump. ~~Two time frames were considered. First,~~ a freeze front analysis ~~was~~ is performed for early times (on the order of seconds ~~or less~~ after vessel failure) to determine the time required to form a plug. ~~The long-term ability of a plug to remain solid was determined using a steady-state analysis.~~ The freeze front analysis is evaluated for three ~~the~~ debris scenarios which envelope the expected debris parameters.

19ED.4.1 Assumptions

The major assumptions invoked in the analyses and their bases follow:

(1) Molten debris enters the channel with negligible superheat.

Molten debris interacts with structural material (steel, concrete, etc.) and the lower drywell environment as it passes from the vessel, contacts the LD floor and spreads to the shield. This interaction depletes the molten debris of any superheat~~, and can result in eutectic formations. The melting temperature of core debris which has undergone little interaction is approximately 2500 K. Significant interaction with the concrete floor reduces the debris melting temperature to approximately 1700 K.~~ To account for uncertainties, the impact of superheat will be considered in the sensitivity study contained in Subsection 19ED.4.5.2

(2) During the freezing process, the temperature profile of the solidified debris rapidly obtains its steady state value.

This assumption introduces little inaccuracy because:

(a) the ~~heat conduction coefficient in the solidified~~ thermal conductivity of the debris is ~~significantly~~ larger than that of the shield material for most debris scenarios, see Subsection 19ED.4.4.

(b) the depth of the ~~solidified~~ debris is ~~considerably less than the height of the shield.~~ only 1 cm; thus, the thermal time constant of the debris in the channel is small compared to the freezing time. This assumption can be checked by comparing the freeze times calculated considering thermal gradients within the debris and the lumped mass analysis used in the superheat study.

semi-infinite solid body only experiencing conduction. The contact temperature between the debris and the channel wall (Reference 19ED-2), assuming semi-infinite bodies, is:

$$T_s = \frac{T_{f,m}\sqrt{(k\rho c)_{cm}} + T_i\sqrt{(k\rho c)_w}}{\sqrt{(k\rho c)_{cm}} + \sqrt{(k\rho c)_w}} \tag{19ED-5}$$

where:

c = specific heat

cm = debris material properties

w = wall material properties

Using the material properties for the wall and the debris contained in Tables 19ED-1 and 19ED-2, respectively

~~Using the debris properties found in Table 19E.2-17, *Important Parameters for Steam Explosion Analysis*, and representative wall properties found in Table 19ED-1~~, the interface temperature is estimated to ~~be 1390 K.~~ be between 1560 and 2180 K.

The debris energy generation density can be found by assuming a decay heat level and a total amount of corium. The density is:

$$\dot{q} = \frac{Q_{dh}\rho_{cm}}{m_{cm}} \tag{19ED-6}$$

where:

Q_{dh} = decay heat level

m_{cm} = total mass of corium, 235 Mg.

text says 1475-2070 K Dm 34

Assuming the decay heat level is approximately 1% of rated power yields:

$$\dot{q} = 1.5 \times 10^6 \text{ MW} / \text{m}^3$$

The two terms inside the brackets in Equation 19ED-4 can now be evaluated. For a channel height of 1 cm ($x_{c,max}$ = 0.5 cm) and a debris melting temperature of 1700 K, these values are:

$$\frac{k_f}{x_c}(T_{f,m} - T_s) = 1.86 \times 10^6 \text{ W} / \text{m}^2$$

(9) The corium shields were assumed to be structurally stable. Structural stability is only an issue during the initial onslaught of debris into the lower drywell. After debris comes into contact with the shields, a crust will form and it will tend to grow in time. Crust formation eliminates buoyancy forces and will hold the individual bricks into place.

19ED.4.2 Initial Freezing of Molten Debris in Channel

If the floor drain sump shield fulfills its design objective, a debris plug will form in the channel before a significant amount of molten corium has a chance to traverse the channel and reach the sump. Molten debris enters the channel at a significantly elevated temperature (1~~7~~800 K to 2500 K) compared to the shield wall (~ 330 K). The walls absorb heat from the debris because of the large temperature difference. Since the debris contains negligible superheat, any heat loss by the debris results in freezing. Freeze fronts start at the channel walls and move toward the center of the channel. ~~The leading edge of the freeze front will stay at the melting temperature of the debris.~~ The freezing process is symmetric about the centerline of the channel because the same amount of heat is transferred through each wall while they are behaving as semi-infinite slabs. The channel walls behave as semi-infinite slabs during the freezing process because the heat conduction rate through the wall material is low compared to the release rate of latent heat. A sketch of the freezing process is shown in Figure 19ED-2.

(1) Freezing Time

ok Did not change

The temperature profile in the ~~crust~~ debris steet (Reference 19ED-1), assuming it quickly reaches its steady state shape, is:

$$T_c(x) = \frac{\dot{q}L_c^2}{2k_f}\left(1 - \frac{x^2}{L_c^2}\right) + \frac{T_s - T_{f,m}}{2}\frac{x}{L_c} + \frac{T_s + T_{f,m}}{2} \qquad \text{(19ED-1)}$$

not changed

where:

$T_c(x)$	=	temperature within the ~~crust~~ debris
x	=	~~crust~~ vertical coordinate measured from the ~~crust~~ steet centerline ok
$\dot{q}$	=	heat density of the ~~crust~~ debris
L_c	=	half thickness of the crust
k_f	=	thermal conductivity of debris
T_s	=	interface temperature between the wall and debris

Insert 19ED.4.3

(4) Mass of Debris Frozen in Channel

The time varying mass of debris freezing in the channel per unit width can be approximated by

$$\dot{m}_{fr} = \frac{d}{d\tau}\left(A' \rho_{cm}\right)$$

where: A' = cross sectional area of frozen debris.

The cross sectional area can be related to the crust thickness by modifying Equation 19ED-8 to account for the variable residency time of the debris at various vertical locations in the channel. This process yields:

$$A' = 2b_0 \int_0^L \sqrt{\tau - y/\bar{v}}\, dy$$

where: L = length from the channel entrance to the leading edge of the debris front

y = vertical coordinate measured from the entrance of the channel.

Evaluating this integral yields:

$$A' = \frac{4}{3} b_0 \bar{v}(\tau)\, \tau^{3/2}.$$

Combining this result with Equations 19ED-25 and 19ED-26 yields:

$$\bar{v}(\tau) = \frac{2}{3} a_0 \sqrt{\tau} - \frac{a_0 b_0}{H_0}\tau - \frac{2 b_0 \bar{v}(\tau)}{H_0}\sqrt{\tau}$$

See 19ED-14

ditto

$$= \frac{\frac{2}{3} a_0 \sqrt{\tau} - \frac{a_0 b_0}{H_0}\tau}{1 + \frac{2 b_0}{H_0}\sqrt{\tau}}.$$

?

The average velocity of debris between the entrance of the channel and the leading edge of molten corium is:

$$\bar{v}(t) = \frac{\int_0^t \ell\, v_o(t)\, dt}{\int_0^t \ell\, dt}$$ (19ED-24)

Evaluating this integral yields:

?

$$\bar{v}(t) = \frac{2}{3} a_o \sqrt{t} - \frac{a_o b_o}{H_o} t - \frac{\dot{m}_{sr}}{\rho_{cm} H_o}$$ (19ED-25)

Not the same as 19ED-2

where:

$$a_o = \frac{4}{5} \sqrt{\frac{2 g \dot{m}_{ves}}{\rho_{cm} A_{ld}}}$$

$$b_o = \frac{5}{3} \sqrt{\frac{2 k_f (T_{f,m} - T_s)}{\rho_{cm} h_{lh}}}$$

This is the time average velocity of the molten debris into the shield channel.

Insert 19ED.4.3

(5) Required Channel Length to Insure Freezing

The channel length, required to ensure a plug forms at the channel entrance before debris spills into the sumps, is:

where t_{freeze} is given by Equation 19ED-9.

$$L_{freeze} = \bar{v}(t_{freeze}) t_{freeze} = a_o t_{freeze}^{3/2} - \frac{a_o b_o}{H_o} t_{freeze}^2$$ (19ED-26)

Inserts 19ED.4.4, 19ED.4.5 and 19ED.4.6

19ED.5 Long-Term ~~Ability of Debris to Remain Solid~~ Capability of the Shield Walls

Initial debris solidification was considered in Subsection 19ED.4. The requirements for keeping the debris in the channel frozen for an extended period of time (at least 24 hours) will be determined in this subsection. The height of the upper shield wall, (above

The RAP activities for important SSCs identified by this Level 2 analysis are given in Table 19K-4.

19K.5 Determination of "Important Structures, Systems and Components" for Seismic Analysis

The seismic analysis considers the potential for core damage from plant damage resulting from a seismic event. The results of the seismic analysis identified key features by consideration of those SSCs important to reactor shutdown or to decay heat removal which could potentially be damaged by seismic action.

The seismic margins analysis calculated high confidence, low probability of failure (HCLPF) accelerations for important accident sequences and classes of accidents. The analysis showed that all SSCs in the analysis have HCLPF equal to or greater than 0.60g, or twice the 0.30g of the safe shutdown earthquake (SSE). Because an important failure mode for beyond design bases earthquakes is the failure of the RHR heat exchanger in such a manner as to drain the suppression pool, the RHR heat exchanger was assigned a HCLPF of 0.7g in the ABWR PRA-based seismic margins analysis.

The two methods that were used to identify important SSCs from the standpoint of seismic analysis are the following:

(1) Identification of the SSCs whose failure would provide the shortest path to core melt in terms of the number of failures required, and comparison of the seismic capacities of those SSCs.

(2) Identification of the most sensitive SSCs in terms of their effect on accident sequence and accident class HCLPFs resulting from variation of component seismic capacities.

in the design certification scope

The primary containment and the reactor building are the Category I structures with the lowest values of HCLPF, but since both have HCLPF greater than 1.1 g no special RAP activities are deemed necessary for these structures. Other SSCs identified by the seismic analysis as being important are as follows:

- The diesel generators, 480 VAC transformers, motor control centers and circuit breakers of the emergency AC Power System
- The batteries and cable trays of the DC Power System
- The heat exchanger of the Residual Heat Removal System
- The SLC tank, valves, and piping and the motor driven pumps of the Standby Liquid Control System
- The valves, piping, and diesel-driven pump of the Fire Water System

19R.4.5

and not expected to open. Even if the suction valve were to open, suppression pool water would fill the ECCS room and flow on to the floor of B2F where it would return to the B3F corridor of the same division via floor drains. The volume of the ECCS Room and the divisional corridor are sufficient to contain the flood water from both the CST and suppression pool.

Floor B2F

Inside secondary containment potential flooding on this floor could occur from the same sources as on B3F (i.e., suppression pool and fire water). A leak in the ECCS chase in each of the divisional valve rooms would cause water to flow down floor drains to the ECCS divisional room on B3F and be processed as previously discussed. Flooding in other areas would be routed through floor drains to the divisional corridor in B3F (Figure 19R-4).

Floors B1F-4F

All inside secondary containment flooding sources on floors B1F-4F would be routed through floor drains to the corridor of B3F and mitigated as discussed above for flooding on B3F.

CUW Line Breaks

The effects of an unisolated reactor water cleanup (CUW) break were analyzed to determine the potential impact on ECCS equipment. The specific effects considered were the possibility of a CUW break rupturing an ECCS wall due to pressure, and the possibility of a CUW break flooding an ECCS room.

The analysis was based on the ABWR secondary subcompartment pressurization analysis (SSPA) model, which postulates a break in each subcompartment through which a CUW high energy line passes. For each postulated break, pressure and temperature transients for each subcompartment were determined using the same methodology as used for compartment pressurization analyses reported in SSAR Subsection 6.2.3.

corridor

Since there are no common walls between ECCS and CUW quadrants, the pressure in the El(-)8200 mm ~~hallway~~ is the only CUW break source that could rupture an ECCS wall. The worst-case CUW break was determined from the SSPA to be a 200 mm double-ended break in the El (-)8200 pump rooms ~~(-)5100~~ mm ~~regenerative heat exchanger room~~. ~~The regenerative heat exchanger is located above the non-regenerative heat exchanger~~, with approxi~~mately 50% of~~ the regenerative heat exchanger room floor area open to the ~~non-regenerative heat exchanger room through the floor grating~~. ~~Therefore, the two rooms were modeled as one node in the analysis.~~ The worst-case break in this analysis was defined as the break which will result in the highest pressure in the El(-)8200 mm ~~Divisional~~ corridor (Figure 19R-3). The entire corridor at elevation (-)8200 mm is modeled as one volume, assuming that divisional separation doors (at this elevation) are open and remain open during the high energy line break events.

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(Figure 5.4-12, sheet 1 of 4)

Break flow is comprised of flow from the reactor side (upstream of break) and from the balance-of-system (BOS) side (downstream of break to check valve). Reactor side break flow is modeled in two distinct phases: a period of unsteady flow called the inventory depletion period followed by steady, critical flow choked at ~~flow orifice~~ venturi FE-001 inside the primary containment. BOS break flow consist. of inventory depletion period flow only since check valves isolate this side of the break from feedwater, the downstream pressure source. The analysis conservatively assumes the complete BOS volume of water, including heat exchangers and filter-demineralizers, will flow out of the break. Steady critical flow is calculated using the Moody Homogenous Equilibrium Critical Flow model.

0.31 kg/cm²g 381

Analysis results showed that the maximum pressure and temperature values for the EL(-)8200 mm corridor during the worst-case CUW break are ~~[illegible] kg/cm²~~ and ~~389~~ K (~~[illegible]~~°C) respectively. These values are below the design pressure and temperature conditions (Tables 6.2-3 and 31.3-15). 107.9

If ~~This analysis~~ conservatively assumed that the Division B corridor contained all the released water. The volume of water released from the worst-case CUW break was determined to be ~~441~~ 459 cubic meters, based on the density of water at ~~389 K (116.1°C)~~ 381 K (107.9). This calculation also assumed that the operator depressurizes the reactor 30 minutes after the break terminating the flood. ~~For the given corridor floor area~~, this volume of water will fill the corridor to a level of approximately 1.4 m. The ECCS watertight doors will ensure that no water enters any of the ECCS rooms. If the break were to occur during shutdown, administrative procedures ensure that at least one ECCS division will be available. ~~In addition, the steam environment caused by the break will not cause failure of any ECCS equipment as they are qualified for this environment. The ECCS equipment referred to here are those equipment and accessories that are part of HPCF, RHR and RCIC systems and are to perform safety functions.~~

Changes in this ¶ were not added

As Not Done

In view of the above, a CUW line break is not expected to cause failure of any ECCS walls. Also, the flood volume will be contained within the corridor~~, and the steam environment will not result in failure of any ECCS equipment~~.

Outside Secondary Containment

Flooding sources outside secondary containment are dominated by fire water leaks. Areas outside secondary containment start at level B1F (i.e., B3F and B2F are entirely within secondary containment.) B1F contains two sump pumps, one each in the Division B and C areas. Floors 1F-4F contain floor drains which all terminate on floor B1F.

The following discussion addresses specific flooding concerns on each floor outside secondary containment.

SCSB Concerns

Primary Containment Pressure Limit (PCPL)

The PCPL is used in the generic Boiling Water Reactor Owners Group (BWROG) Emergency Procedure Guidelines (EPGs) to define the pressure at which containment venting should occur to prevent uncontrolled containment failure. The PCPL is defined as the lowest pressure of the following: (1) pressure capability of the containment, (2) maximum containment pressure for vent valves to open and close, (3) maximum containment pressure safety relief valves can open and remain open, and (4) maximum containment pressure for RPV vent valves to open and close for containment flooding. Within Revision 4 of the BWROG EPGs, which were reviewed and approved by the NRC in 1988, containment venting was to occur prior to reaching the PCPL.

Within Amendment 33 of the GE ABWR SSAR, two different values (80 psig and 90 psig) for the PCPL were stated. SCSB identified this as discrepancy 1A in our review of Amendment 33 in January 1994. In the December 1993 advance copy of the final safety evaluation report, the staff concluded that the COPS actuation setpoint of 90 psig was acceptable since it was the PCPL. Since that time, GE has concluded that the correct PCPL is 80 psig, limited by the operability of the SRVs. With COPS actuation above the PCPL, a depressurized primary system at the time of containment depressurization is not assured.

A PCPL of 80 psig, based on SRV operability, and COPS actuation of 90 psig has significant ramifications including:

1. The ABWR reactor vessel depressurization system may not meet the criteria of SECY-90-016 which require a reliable depressurization system to ensure reactor vessel failure at low pressure in a core melt scenario to prevent high pressure melt ejection.

2. The ABWR EPGs are a departure from the BWROG EPGs approved by the NRC in that venting is no longer performed prior to reaching the PCPL.

3. The fraction of reactor vessel failures that occur at high pressure may be considerably higher than modelled in the PRA. The core damage frequency would also be higher since core injection via ACIWA would be terminated upon reclosure of the SRVs.

Resolution: In a May 4, 1994, conference call, GE agreed to re-evaluate the PCPL and determine the impact on the SRVs of raising the accumulator pressure to ensure SRV operability beyond COPS actuation. SCSB will continue discussions with GE concerning this issue.

Containment Sprays

The ABWR containment spray system consists of the residual heat removal (RHR) system pumping suppression pool water through RHR heat exchangers to the wetwell and drywell spray sparger. Within Amendment 33, the use of the containment spray system was unclear. In certain areas, GE stated that the containment spray system could only be operated in the wetwell/drywell spray mode combined. In other areas GE discussed the independent operation of wetwell or drywell sprays. SCSB believed that it was a combined system and performed our evaluations accordingly.

Within the following areas of the FSER, SCSB based its safety findings on the understanding the containment spray system can only be operated in unison (wetwell and drywell together):

- FSER 6.2.1.8: 'Wetwell spray cannot be operated in isolation. Drywell spray and wetwell spray are required to operate in unison to handle the flow rate of the RHR pumps. An orifice is included in the wetwell spray line to limit the flow to the stated amount. Wetwell spray flow is balanced by the orifice to provide (1) enough flow to mitigate pressurization of the wetwell due to steam bypass and (2) not too much flow to exceed the negative design pressure of the wetwell as discussed in Section 6.2.1.1.4 of the ABWR SSAR."

- FSER 18.8.3: "The DSIL curve used in the ABWR EPGs differs from the one used in Revision 4 of the BWROG EPGs. The curve to the right of the peak has been eliminated for the ABWR because the possibility of a large pressure differential between the wetwell and drywell at the onset of drywell sprays is not likely since the drywell and wetwell sprays actuate simultaneously in ABWR. . . .The staff finds the revised DSIL curve acceptable because the drywell and wetwell sprays actuate simultaneously in the ABWR which eliminates the possibility of a significant pressure differential between the wetwell and the drywell at the onset of drywell sprays."

GE revised the drywell spray initiation limit curve of the EPGs based on combined spray flow operation to delete restrictions on spray operation which could result in excessive negative differential pressures. The basis provided by GE for this revision was the combined operation of wetwell and drywell sprays. However, independent operation is possible which would not be covered within the revised EPGs.

In addition, GE evaluated sequences that could result in a negative containment pressure differential - drywell below wetwell, drywell below reactor building, and wetwell below reactor building. The results of the sequences evaluated were used in determining the sizing of the wetwell to drywell vacuum breakers and concluding that reactor building to wetwell vacuum breakers are not needed. However, the sequences evaluated did not look at the case of inadvertent drywell spray actuation following a LOCA. The following is a summary of the sequences evaluated by GE:

Negative Containment Design Pressures

- WW-DW 13.7 kPaD
- RB-DW 13.7 kPaD
- RB-WW 13.7 kPaD

Wetwell Greater than Drywell/Reactor Bldg. Greater than Drywell

Concern: containment vessel/liner, diaphragm floor, pedestal, and lower drywell access tunnels

Scenarios Evaluated:

1. Inadvertent DW spray during normal operation

2. DW depressurized following LOCA (FWLB)

   - severest pressure transient in DW
   - used for sizing WW to DW vacuum breakers
   - cold ECCS flow through the break
   - WW to DW pressure 9.8 kPaD
   - WW-DW vacuum breakers needed

3. DW/WW spray following LOCA

   - WW/DW to RB pressure 5.9 kPaG
   - no RB-WW vacuum breakers needed

Reactor Bldg. Greater than Wetwell

Concern: containment vessel/liner

Scenarios Evaluated:

1. DW/WW spray during normal operation

   - 2 cases 6.9 and 11.8 kPaG

2. WW spray following SORV

   - 11.8 kPaG

3. DW/WW spray following LOCA

   - WW/DW to RB pressure 5.9 kPaG

Resolution: GE should assess the impact of drywell spray actuation following a LOCA and ensure that the bounding scenario has been evaluated. GE should re-assess the drywell spray initiation limit curve and determine the impact of drywell spray actuation on the differential pressure capability of the containment.