GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

January 11, 1991

MFN No. 004-91
Docket No. STN 50-605
EEN-9105

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Charles L. Miller, Director
Standardization and Non-Power Reactor Project Directorate

Subject: GE Responses to GE/NRC Conference Calls on Reactor Systems, SER Input for ABWR SSAR Chapters 4 5, 6, 9 and 15

Reference: GE Responses (Proprietary Information) to GE/NRC Conference Calls on Reactor Systems, SER Input for ABWR SSAR Chapters 4 5, 6, 9 and 15, MFN No. 005-91, dated January 11, 1991

Enclosed are thirty four (34) copies of the GE responses to the discussion items of the subject conference calls made on December 4, 1990 and December 6, 1990.

Responses to discussion items 1, 5, and 6 contain information that is designated as General Electric Company proprietary information and is being submitted under separate cover (Reference).

It is intended that GE will amend the SSAR, where appropriate, with these responses in a future amendment.

Sincerely,

R C Stirn

R.C. Stirn, Acting Manager
Regulatory and Analysis Services
M/C 382, (408) 925-6948

cc: F. A. Ross (DOE)
D. C. Scaletti (NRC)
G. Thomas (NRC)
D. R. Wilkins (GE)
J. F. Quirk (GE)

D028
/1

RESPONSES TO DISCUSSION ITEMS REQUIRING GE ACTION

DISCUSSION ITEM 1

Markup new ABWR SSAR Appendix 4B to be consistent with GESTAR II Amendment 22. Also, provide a statement indicating that Appendix 4B is to be used if analysis is required.

RESPONSE 1

Proprietary information provided under separate cover.

DISCUSSION ITEM 4

Revise Section 4.4 to include a loose-parts monitoring system to be supplied by the applicant referencing the ABWR design . This revision is to include a statement why it is advantageous to delay design of the loose-parts monitoring system until detailed design or the reactor system has been completed and equipment vendors selected.

RESPONSE 4

See attached markup.

DISCUSSION ITEM 5

Revise ABWR SSAR Subsection 15.2 to address the failure mode and effects of slow air system failure or dirty air in the instrument air system.

RESPONSE 5

Proprietary information provided under separate cover.

DISCUSSION ITEM 6

Submit detailed information including drawings of the HCU.

RESPONSE 6

Proprietary information provided under separate cover.

DISCUSSION ITEM 7

Revise Subsection 5.4.7.1.3 to reflect compliance with RSB BTP 5-1.

RESPONSE 7

See attached markup.

DISCUSSION ITEM 11

Provide calculations to substantiate the adequacy of the 100 gpm borated solution in meeting 10CFR50.62(a)(4).

REPONSE 11

As approved and documented in the Safety Evaluation of Topical Report (NEDE-31096-P-A) "Anticipated Transient Without Scram; Response to ATWS Rule, 10CFR50.62" (October 1986), the equivalency requirement can be demonstrated if the following relationship is satisified:

$$\frac{Q}{86} \times \frac{M251}{M} \times \frac{C}{13} \times \frac{E}{19.8} \geq 1$$

where

Q = expected SLCS flow rate (gpm)

M = mass of water in the reactor vessel and recirculation system a hot rated condition (lbs)

C = sodium pentaborate solution concentration (weight per cent)

E = B^{10} isotope enrichment (19.8% for natural boron), atom percent

Values of M251 may vary somewhat depending on the design, (e.g., M251 for BWR/3/4 = 628,300 lbs, M251 for BWR/5 = 614,300 lbs, and M251 for BWR/6 = 615,100 lbs), or

$$Q \geq 86 \times \frac{M}{M251} \times \frac{13}{C} \times \frac{19.8}{E}$$

For ABWR,

M = 674,100 lbs

C = 13

E = 19.8

and using M251 = 615,100 lbs, and obtains

$Q \geq 94.2$ gpm.

Therefore, the 100 gpm capacity selected for the ABWR satisifies the requirment.

DISCUSSION ITEM 14

Review and update the GE Feburary 21, 1980 responses to the Michelson concerns for applicability to ABWR.

RESPONSE 14

See attached markup.

A

4.4.~~3~~ 4 Interfaces

4.4.~~3~~ 4.1 Power Flow Operating Map

The specific power flow operating map to be used at the plant will be provided by the utility to the USNRC for information.

4.4.~~3~~ 4.2 Thermal Limits

The thermal limits for the core loading at the plant will be provided by the utility to the USNRC for information.

B

INSERTS FOR DISCUSSION ITEM 4

(A) 4.4.3 Loose-Parts Monitoring System

The applicant referencing the ABWR design shall provide a loose-parts monitoring system on the reactor pressure vessel, and implement a loose-parts detection program which conforms to the guidelines of the regulatory position contained in Regulatory Guide 1.133. The design of the loose-parts monitoring system is deferred so that it may be defined utilizing commerically available components at the time of construction in a system and it can reflect applicant preference and perhaps experiencd may be best satisified. See Subsection 4.4.4.3 for interface requirements.

(B) 4.4.4.3 Loose-Parts Monitoring System

The applicant referencing the ABWR design will provide a loose-parts monitoring system and implement a loose-parts detection program (See Subsection 4.4.3).

MARKUP FOR DISCUSSION ITEM 7

directly into the reactor pressure vessel to the drywell spray header degraded plant conditions when AC power is not available from either onsite or offsite sources. The RHR provides the piping and valves which connect the FPS piping with the RHR loop C pump discharge piping. The manual valves in this line permit adding water from the FPS to the RHR system if the RHR is not operable. The primary means for supplying water through this connection is by use of the diesel-driven pump in the FPS. A backup to this pump is provided by a connection on the outside of the reactor building which allows hookup of the FPS to a fire truck pump.

The vessel injection mode is intended to prevent core damage during station blackout after RCIC has stopped operating, and to provide an in-vessel core melt prevention mechanism during a severe accident condition. If the AC-independent water addition mode is not actuated in time to prevent core damage, core melting and vessel failure, then it covers the corium in the lower drywell when initiated and adds water to containment, thereby slowing the pressure rise.

The drywell spray mode prevents high gas temperatures in the upper drywell and adds additional water to the containment, which increases the containment thermal mass and slows the pressurization rate. Additionally, the drywell spray provides fission product scrubbing to reduce fission product release in the event of failure of the drywell head.

Operation of the AC-independent water addition mode is entirely manual. All of the valves which must be opened or closed during fire water addition are located within the same ECCS valve room. The connection to add water using a fire truck pump is located outside the reactor building at grade level.

5.4.7.1.2 Design Basis for Isolation of RHR System from Reactor Coolant System

The low pressure portions of the RHR system are isolated from full reactor pressure whenever the primary system pressure is above the RHR system design pressure. (See Subsection 5.4.7.1.3 for further details.) In addition, automatic Isolation occurs for reasons of maintaining water inventory which are unrelated to line pressure rating. A low water level signal closes the RHR containment isolation valves that are provided for the shutdown cooling suction. Subsection 5.2.5 provides an explanation of the leak detection system and the isolation signals; see Subsection 5.2.5.2.1 (12) and Table 5.2-6.

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves which open on low mainline flow and close on high mainline flow.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR system are sized on ~~one of three basis:~~ the basis of

(1) thermal relief, ~~only,~~ and

(2) valve bypass leakage only.

(3) control valve failure and the subsequent uncontrolled flow which results.

Redundant interlocks prevent opening valves to the low pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure.

A

In addition, a high pressure check valve will close to prevent reverse flow if the pressure should increase. Relief valves in the discharge piping are sized to account for leakage past the check valve.

5.4.7.1.4 Design Basis With Respect to General Design Criterion 5

The RHR system for this unit does not share equipment or structures with any other nuclear unit.

5.4.7.1.5 Design Basis for Reliability and Operability

The design basis for the shutdown cooling mode of the RHR System is that this mode is controlled by the operator from the control room. The only operations performed outside of the control room for a normal shutdown are manual operation of local flushing water admission valves, which are the means of providing clean water to the shutdown portions of the RHR system.

Three separate shutdown cooling loops are provided; and although the three loops are required for shutdown under normal circumstances, the reactor coolant can be brought to 100°C in less than 36 hours with only two loops in operation. The RHR system is part of the ECCS and therefore is required to be designed with redundancy, piping protection, power separation, etc., as required of such systems. (See Section 6.3 for an explanation of the design bases for ECCS Systems.)

Shutdown suction and discharge valves are required to be powered from both offsite and standby emergency power for purposes of isolation and shutdown following a loss of offsite power.

5.4.7.1.6 Design Basis for Protection from Physical Damage

The design basis for protection from physical damage, such as internally generated missiles, pipe break, seismic effects, and fires, are discussed in Sections 3.5, 3.6, 3.7, and Subsection 9.5.1

5.4.7.2 Systems Design

5.4.7.2.1 System Diagrams

All of the components of the RHR system are shown in the P&ID (Figure 5.4-10). A description of the controls and instrumentation is presented in Subsection 7.3.1.1.1 emergency core cooling systems control and instrumentation.

Figure 5.4-11 is the RHR process diagram and data. All of the sizing modes of the system are shown in the process data. The interlock block diagram (IBD) for the RHR system is provided in Section 7.3.

Interlocks are provided to prevent: (1) drawing vessel water to the suppression pool, (2) opening vessel suction valves above the suction lines or the discharge line design pressure, (3) inadvertent opening of drywell spray valves during RHR operation where the injection valve to the reactor is open and when drywell pressure is not high enough to require the drywell spray for pressure reduction, and (4) pump start when suction valve(s) are not open. A description of the RHR system logic (i.e., interlocks, permissives) is presented in Table 5.4-3.

5.4.7.2.2 Equipment and Component Description

(1) System Main pumps

The following are system performance requirements the main pumps must satisfy. The pump equipment performance requirements include additional margins so that the system performance requirements can be achieved. These margins are standard GE equipment specification practice and are included in procurement specifications for flow and pressure measuring accuracy and for power source frequency variation.

Number of Pumps	3
Pump type	Centrifugal
Drive unit type	Motor

INSERT FOR DISCUSSION ITEM 7

Overpressure protection is achieved during system operation when the system is not isolated from the reactor coolant pressure. The RHR system is operational and not isolated from the reactor coolant system only when the reactor is depressurized. Two modes of operation are applicable; the flooder move and the shutdown cooling mode. For the flooder mode, the injection valve opens through interlocks only for reactor pressures less than approximately 500 psig. For the shutdown cooling mode, the suction valves can be opened through interlocks only for reactor pressures less than approximately 135 psig. Once the system is operating in these lower pressure modes, events are not expected that would cause the pressure to increase. If for some unlikely event the pressure would increase, the pressure interlocks that allowed the valves to initially open would cause the valves to close on increasing pressure. The RHR system piping would then be protected from overpressure. The valves close at low pressure, and the rate of pressure increase would be low. During the time period while the valves are closing at these low pressure conditions, the RHR system design and margins that satisfy the interfacing system LOCA provide ample operpressure protection.

single failure, the reactor core remains covered with water until stable conditions are achieved. Furthermore, even with more degraded conditions involving a stuck-open relief valve in addition to the worst transient (loss of feedwater) and worst single failure (of high pressure core spray), studies show (NEDO-24708, March 31, 1980) that the core remains covered and adequate core cooling is available during the whole course of the transient. The conclusion is applicable to the ABWR. Since the ABWR has more high pressure make-up systems (2HPCFs and 1 RCIC), the core covering is further assured.

Other discussions of transients with single failure is presented in the response to NRC Question 440.111.

1A.2.33.2 Evaluate Depressurization other than Full ADS [II.K.3 (45)]

NRC Position

Provide an evaluation of depressurization methods other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown. (Applicable to BWR's only)

Response

This response is provided in Subsection 19A.2.11

1A.2.33.3 Responding to Michelson Concerns [II.K.3 (46)]

NRC Position

General Electric should provide a response to the Michelson concerns as they relate to boiling water reactors.

Clarification

General Electric provided a response to the Michelson concerns as they relate to boiling water reactors by letter dated February 21, 1980. Licensees and applicants should assess applicability and adequacy of this response to their plants.

Response

All of the generic February 21, 1980 GE responses were reviewed and updated ~~are applicable to the ABWR design and are adequate in terms of a response to the Michelson concerns~~ for the ABWR Standard Plant. The specific responses are provided in Table 1A.2-1.

1A.2.34 Primary Coolant Sources Outside Containment Structure [III.D.1.1(1)]

NRC Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with systems in operation and report them to the NRC.

(2) Continuing Leak Reduction--establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This

TABLE 1A.2-1
RESPONSES TO QUESTIONS POSED BY MR. C. MICHELSON [II.K.3(46)]

QUESTION 1
Pressurizer level is an incorrect measure of primary coolant inventory.

RESPONSE 1
BWRs do not have pressurizers. BWRs measure primary coolant inventory directly using differential pressure sensors attached to the reactor vessel. Thus, this concern does not apply to ABWR.

QUESTION 2
The isolation of small breaks (e.g., letdown line; PORV) not addressed or analyzed.

RESPONSE 2
Automatic isolation only occurs for breaks outside the containment. Such breaks are addressed in Section 3.1.1.1.2 of NEDO-24708. It was shown that if the high pressure systems are available no operator actions are required. If it is assumed that all high pressure systems fail, the operator must manually depressurize to allow the low pressure systems to inject and maintain vessel water level. Analyses in Section 3.5.2.1 of NEDO-24708 show that the operator has sufficient information and time to perform these manual actions. The necessary manual actions have been included in the operator guidelines for small break accidents.

QUESTION 3
Pressure boundary damage due to loadings from 1) bubble collapse in subcooled liquid and 2) injection of ECC water in steam-filled pipes.

RESPONSE 3
The BWR has no geometry equivalent to that identified in Michelson's report on B&W reactors relative to bubble collapse (steam bubbling upward through the pressurizer surge line and pressurizer). Thus the first concern in not applicable to ABWR.

ECC injection in the ABWR at high pressure is either directly into the reactor vessel through water filled lines (RHR-B+C;HPCF-B+C) or into the feedwater lines (RHR-A;RCIC). The feedwater lines are normally filled with relatively cold liquid (420°F or less). ECCS injection in the ABWR at low pressure is either directly into the reactor vessel or into the feedwater lines. Thus the second concern is not applicable to the ABWR.

QUESTION 4
In determining need for steam generators to remove decay heat, consider that break flow enthalpy is not core exit enthalpy.

RESPONSE 4
BWRs do not use steam generators to remove decay heat, so this concern does not apply to ABWR.

QUESTION 5

Are sources of auxiliary feedwater adequate in the event of a delay in cooldown subsequent to a small LOCA?

RESPONSE 5

BWRs do not need feedwater to remove heat from the reactor following a LOCA, whether the subsequent cooldown is delayed or not. Therefore, this concern is not applicable to ABWR. BWRs have a closed cooling system in which vessel water flows out the postulated break to the suppression pool. The suppression pool is cooled and water is pumped back to the vessel with ECCS pumps.

QUESTION 6

Is the recirculation mode of operation of the HPCI pumps at high pressure an established design requirement?

RESPONSE 6

The high-pressure injection systems utilized in the ABWR are the Reactor Core Isolation Cooling (RCIC) and High Pressure Core Flooder (HPCF).

The RCIC and HPCF systems normally take suction from the condensate storage tank and have an alternate suction source from the suppression pool. A recirculation mode of operation of these systems is established when the system suction is from the suppression pool. Following a LOCA when system suction is from the suppression pool, water injected into the reactor is discharged through the break and flows back to the suppression pool forming a closed recirculation loop.

Other recirculation modes include test modes (e.g., suction from and discharge to the suppression pool) and system operation on low flow bypass with discharge to the suppression pool.

All of these modes are established design requirements.

QUESTION 7

Are the HPCI pumps and RHR pumps run simultaneously? Do they share common piping?/suction? If so, is the system properly designed to accommodate this mode of operation (i.e., are any NPSH requirements violated, etc...?)

RESPONSE 7

For ABWR the high-pressure injection systems (RCIC/HPCF) do not share any common suction piping with the low pressure RHR and they can operate simultaneously with this low pressure system.

The RCIC and HPCF systems share a common suction line from the condensate storage tank. The RHR shutdown cooling operating mode does not share any common suction piping with the RCIC or HPCF systems. It is an established design requirement to size the suction piping, including shared piping, such that adequate NPSH is available to RCIC and HPCF pumps for all simultaneous operating modes of these systems.

Pre-operational and/or start-up tests are conducted that demonstrate that the NPSH requirements are met.

QUESTION 8
Mechanical effects of slug flow on steam generator tubes needs to be addressed (transitioning from solid natural circulation to reflux boiling and back to solid natural circulation may cause slug flow in the hot leg pipes.)

RESPONSE 8
BWRs do not have steam generators so this concern does not apply to ABWR.

QUESTION 9
Is there minimum flow protection for the HPCI pumps during the recirculation mode of operation?

RESPONSE 9
For the ABWR, the RCIC, RHR, and HPCF pumps all contain valves, piping, and automatic logic that bypasses flow to the suppression pool as required to provide minimum flow protection for all design basis operating modes of the systems.

QUESTION 10
The effect of the accumulators dumping during small break LOCAs is not taken into account.

RESPONSE 10
BWRs do not use accumulators to mitigate LOCAs. Therefore, this concern does not apply to ABWR.

QUESTION 11
What is the impact of continued running of the RC pumps during a small LOCA?

RESPONSE 11
The impact of continued running of the recirculation pumps has been addressed in Sections 3.3.2.2, 3.3.2.3, and Section 3.5.2.1.5.1 of NEDO-24708. The conclusions were that continued running of the recirculation pumps results in little change in the time available for operator actions and does not significantly change the overall system response.

QUESTION 12
During a small break LOCA in which offsite power is lost, the possibility and impact of pump seal damage and leakage has not been evaluated.

RESPONSE 12
The RCIC, HPCF, and RHR pumps are provided with mechanical seals. These seals are cooled by the pump primary process water. No external cooling from auxiliary support systems, such as site service water or room air coolers, is required for RCIC pump seals. The HPCF and RHR seals are cooled by connections to the three separate divisions of the Reactor Building Cooling Water (RCW) System to protect against single failures. RHR Divisions A, B and C, and HPCF-B and C are connected to their corresponding RCW divisions. If offsite power is lost, on site diesel generations maintain the RCW three divisional function. These types of seals have demonstrated (in nuclear and other applications) their capability to operate for extended period of time at temperatures in excess of those expected following a LOCA.

Should seal failure occur it can be detected by room sump high level alarms. The RCIC, HPCF, and RHR individual pumps are arranged, and motor operated valves provided, so that a pump with a failed seal can be shutdown and isolated without affecting the proper operation of the other redundant pumps/systems.

Considering the low probability of seal failure during a LOCA, the fact that a pump with a failed seal can be isolated without affecting other redundant equipment, and the substantial redundance provided in the BWR emergency cooling systems, pump seal failure is not considered a significant concern.

<u>QUESTION 13</u>
During transitioning from solid natural circulation to reflux boiling and back again, the vessel level will be unknown to the operators, and emergency procedures and operator training may be inadequate. This needs to be addressed and evaluated.

<u>RESPONSE 13</u>
There is no similar transition in the BWR case. In addition, the BWR has water level measurement within the vessel and the indication of the water level is incorporated into the operator guidelines. Consequently this concern does not apply to ABWR.

<u>QUESTION 14</u>
The effect of non-condensable gas accumulation in the steam generators and its possible disruption of decay heat removal by natural circulation needs to be addressed.

<u>RESPONSE 14</u>
The effect of non-condensable gas accumulation is addressed in Section 3.3.1.8.2 of NEDO-24708. For a BWR, vapor is present in the core during both normal operation and natural circulation conditions. Non-condensables may change the composition of the vapor but would have an insignificant effect on the natural or forced circulation itself, since the non-condensables would rise with the steam to the top of the vessel after leaving the steam separators.

<u>CONCERN 15</u>
Delayed cooldown following a small break LOCA could raid the containment pressure and activate the containment spray system.

<u>RESPONSE 15</u>
The ABWR containment spray system is manually initiated. All essential equipment inside the containment is required to be qualified for the environmental conditions resulting from the initiation of the containment spray system.

QUESTION 16*

This concern relates to the possibility that an operator may be included and perhaps even trained to isolate, where possible, a pipe break LOCA without realizing that it might be an unsafe action leading to high pressure, and short-term core bakeout. For example, if a BWR should experience a LOCA from a pressure boundary failure somewhere between the pump suction and discharge valve for either reactor recirculation pump, it would be possible for the operator to close these valves following the reactor blowdown to repressurize the reactor coolant system. Before such isolation should be permitted, it is first necessary to show by an appropriate analysis that the high pressure ECCS is adequate to reflood the uncovered core without assistance from the low pressure ECCS which can no longer delivery flow because of the repressurization. Otherwise, such isolation action should be explicitly forbidden in the emergency operating instructions.

RESPONSE 16

The ABWR does not have recirculation lines. However, there are other systems where it is possible for the operator to close these valves following the reactor blowdown to low pressure and thereby isolate the break. An example of this would be a break in the Reactor Water Cleanup piping between the shutdown suction line valve and the containment boundary. In Reference 2, the NRC concluded based on information provided by GE that break isolation is not a problem.

In order for the reactor vessel to repressurize following isolation of a line break, the isolation would have to occur before initiation of ADS due to a high drywell pressure in concurrence with low water level 1 condition. Isolation of a line break prior to obtaining a high drywell pressure signal might occur for very small breaks (area << 0.01 ft^2) which may require several hundred seconds following the break to reach the high drywell pressure set point. In this case it has been shown in Reference 3 that the high pressure systems (RCIC, HPCF and feedwater) are sufficient to maintain the water level above the top of the core.

If isolation of the break were to occur prior to reaching level 1 but after the high drywell pressure signal, the vessel would pressurize to the SRV set point following isolation of the main steam lines and then oscillate as the SRVs cycle open and closed. If no high pressure systems were available, the loss of mass out the SRVs would cause the level to continue dropping and result in automatic ADS actuation shortly after reaching level 1. This would depressurize the vessel and allow the low pressure systems to begin injecting. This capability was demonstrated in NEDO-24708, in addition, explicitly provide for manual depressurization in the event of low reactor water level with high pressure systems unable to maintain level for any reason.

---

*Excerpt from Reference 1.

In summary, in order to repressurize the vessel following break isolation, the isolation would have to occur prior to ADS blowdown. For these cases, high pressure systems would maintain inventory. If no high pressure system was available, the low pressure systems would control the vessel water level following automatic or manual vessel depressurization.

REFERENCES:

1. Memo, C. Michaelson to D. Okrent, "Possible Incorrect Operator Action Such as Pipe Break Isolation," June 4, 1979.

2. Letter, D. G. Eisenhut to R. L. Gridley, "Potential for Break Isolation and Resulting GE-Recommended BWR/3 ECCS Modifications," June 14, 1978.

3. "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," NEDO-24708, August 1979.