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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 342 Related to ESBWR Design Certification Application ESBWR RAI Numbers 17.4-51 and 19.1-144 S04**

The purpose of this letter is to submit the GE Hitachi Nuclear Energy (GEH) response to the U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI) dated May 15, 2009 (Reference 1).

The previous RAIs and responses to RAI 19.1-144 are in references 2 through 9. The responses to RAI Numbers 17.4-51 and 19.1-144 S04 are in Enclosure 1.

If you have any questions or require additional information, please contact me.

Sincerely,

Lee F. Dougherty for

Richard E. Kingston
Vice President, ESBWR Licensing

References:

1. MFN 09-332, Letter from U.S. Nuclear Regulatory Commission to Jerald G. Head, GEH, *Request For Additional Information Letter No. 342 Related To ESBWR Design Certification Application*, dated May 14, 2009.

2. MFN 06-442, Supplement 3, *Response to Portion of NRC Request for Additional Information Letter No. 272 Related to E5BWR Design Certification Application E5BWR RAI Number 19.1-144 S03*, dated February 9, 2009

3. MFN 08-890, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 272 Related To ESBWR Design Certification Application*, dated November 5, 2008.

4. MFN 08-552, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 214 Related To ESBWR Design Certification Application*, dated June 25, 2008.

5. MFN 06-442, Supplement 2, *Response to Portion of NRC Request for Additional Information Letter No. 214 Related to E5BWR Design Certification Application E5BWR RAI Numbers 19.1.0-4502 and 19.1-144 502*, dated October 1, 2008

6. MFN 07-555, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, GEH, *Request For Additional Information Letter No. 109 Related To ESBWR Design Certification Application*, dated October 12, 2008.

7. MFN 07-423 S04, *Response to Portion of NRC Request for Additional Information Letter No.109 Related to E5BWR Design Certification Application, RAI Number 19.1-144 S01*, dated April 2, 2008.

8. MFN 07-104, Letter from U.S. Nuclear Regulatory Commission to David H. Hinds, GEH, *Request For Additional Information Letter No. 91 For The ESBWR Design Certification Application, dated January 31, 2007.*

9. MFN 07-423, *Response to Portion of NRC Request for Additional Information Letter No. 91 Related to ESBWR Design Certification Application ESBWR Probabilistic Risk Assessment RAI Numbers 19.1-117 through 19.1-133, 19.1-140, 19.1-142, 19.1-144, 19.1-148, 19.2-69 through 19.2-74 and 19.2-76 through 79,* dated August 13, 2007.

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 342 Related to ESBWR Design Certification Application Probabilistic Risk Assessment RAI Numbers 17.4-51 and 19.1-144 S04

2. DCD Revision 6 and NEDO-33201 Revision 3 Markups

cc:	AE Cubbage	USNRC (with enclosure)	
	JG Head	GEH/Wilmington (with enclosure)	
	DH Hinds	GEH/Wilmington (with enclosure)	
	eDRFSection	0000-0102-3037	RAI 17.4-51
		0000-0096-5462	RAI 19.1-144S04

Enclosure 1

MFN 06-442, Supplement 4

Response to NRC Request for

Additional Information Letter No. 342

Related to ESBWR Design Certification Application

ESBWR Probabilistic Risk Assessment

RAI Numbers 17.4-51 and 19.1-144 S04

[1] ***Original Responses previously submitted under MFN 06-442 and MFN 06-442 S01, S02 and S03 are included to provide historical continuity during review.***

NRC RAI 17.4-51

Regulatory Guide 1.206, Section C.III.1, Subsection C.I.17.4.4, states that the combined license (COL) applicant should describe the quality elements (organization, design control, procedures and instructions, records, corrective action, and audit plans) for developing and implementing the design reliability assurance program (D-RAP) in Chapter 17 of the final safety analysis report (FSAR), in accordance with the provisions in SRP Section 17.4. Section 17.4.5 ("GEH Organization for D-RAP") of the ESBWR Design Control Document (DCD) describes the quality elements for developing and implementing the D-RAP that GEH imposed during the design certification phase. While the quality elements for developing and implementing the D-RAP that are applied by a COL applicant referencing the ESBWR DCD may be similar to that described in Section 17.4.5 of the ESBWR DCD, the COL applicant should impose its own quality elements for developing and implementing D-RAP.

The staff requests that GEH revise COL information item 17.4-1-A to include the description of the quality elements for developing and implementing the D-RAP (i.e., organization, design control, procedures and instructions, records, corrective action, and audit plans) that will be applied by the COL applicant prior to the initial fuel load. This can be accomplished by adding the following statement at the end of the text in COL information item 17.4-1-A:

> *"and describe the quality elements for developing and implementing the D-RAP (i.e., organization, design control, procedures and instructions, records, corrective action, and audit plans) that will be applied prior to the initial fuel load."*

GEH Response

DCD Tier 2 Section 17.4 will be revised to include the requested statement.

DCD Impact

DCD Tier 2 Section 17.4 will be revised as shown in the attached mark-up.

NRC RAI 19.1-144

Please provide calculations to demonstrate that short term and long term core cooling can be provided by isolation condenser system operation following an extended loss of the RWCU/SDC function from a cold shutdown condition. This information will assist the staff to address thermal-hydraulic uncertainty in the ESBWR passive design regarding shutdown success criteria.

GEH Response

The Technical Specifications for the ICS system require at least two of the four loops are operational during Mode 5 (Cold Shutdown). Additionally, the ICS instrumentation & actuation logic are required to be operable in Mode 5 (DCD LCO 3.3.5.3 and 3.3.5.4). ICS is only credited for Mode 5 during shutdown. It will not function during Mode 6 with the reactor vessel head removed.

Loss of RWCU/SDC during shutdown would cause RPV pressure and temperature to both increase to near or above Mode 1 conditions. It is assumed that following loss of RWCU/SDC, the ICS would respond just at it would to a transient during power operations. The primary difference is that only two ICS loops are available and both are needed to meet the success criteria.

Two ICS heat exchangers can remove 1.5% of decay heat. This level of decay heat occurs between 2000 and 4000 seconds following shutdown. Mode 5 (Cold Shutdown) will not occur until well after this time. Two functioning Isolation Condensers can meet the long term and short-term core cooling needs during Cold Shutdown.

DCD/NEDO-33201 Impact

No DCD changes will be made in response to this RAI.

NEDO-33201 Rev 2 Chapter 16 will be revised as described above.

<u>NRC RAI 19.1-144 S01</u>

The NRC staff has reviewed GEH's response to RAI 19.1-144. This RAI requested GEH to provide calculations that demonstrate short term and long term core cooling using the ICS following an extended loss of the RWCU/SDC function from a cold shutdown condition. GEH did not provide the requested calculation. Instead, GEH responded that following loss of RWCU/SDC that ICS would respond just as it would to a transient during power operations. The staff is aware that the heat removal capability of the ICS is credited in the simulations of an ESBWR LOCA event in which there are non-condensible gases present. The NRC staff noted that comparison of the TRACG results to the PANTHERS data show that TRACG does not adequately model the timing of the noncondensible gas transport in the IC. The NRC staff is concerned about the impact of non-condensible gases given ICS operation from a cold shutdown condition. Also, the NRC staff seeks information on the anticipated RCS pressures and temperatures at which ICS operation can provide sustained core cooling. Therefore, as requested in RAI 19.1-144, please provide a TRACG run that demonstrates short term and long term core cooling using the ICS following an extended loss of the RWCU/SDC function from a cold shutdown condition.

<u>GEH Response</u>

The performance of the ICS system has been evaluated for the initial condition of the reactor in Mode 1. The Mode 1 initial condition is bounding for operation of the ICS system as compared to starting from Mode 5 because the level of decay heat is lower for the Mode 5 condition (e.g. ~ 8 hours after as opposed to very soon after reactor scram). The primary difference between these two initial conditions is the time delay involved with heating and pressurizing the reactor vessel when the initial condition is Mode 5. If non-condensable gases accumulate in the IC such that the heat transfer drops below the decay heat of the reactor, the reactor pressure would increase to the setpoint for opening of the ICS vent line. The ICS venting process is the same irrespective of the initial condition operating mode. No specific calculations have been performed for the ICS system with an initial condition of the reactor in Mode 5 since it is bounded by operation from Mode 1. DCD Table 15.1-5 "NSOA System Event Matrix" identifies the NSOA events in which the ICS operates with Mode 1 initial conditions.

<u>DCD Impact</u>

No DCD changes will be made in response to this RAI.

NRC RAI 19.1-144 S02

The staff reviewed GEH's response to RAI 19.1-144 S01. The NRC staff is concerned about the impact of non-condensable gases given Isolation Condenser system (ICS) operation from a cold shutdown condition. The NRC staff also requested a calculation to demonstrate that short term and long term core cooling can be provided by ICS operation following an extended loss of the RWCU/SDC function from a cold shutdown condition. Successful, automatic, IC operation significantly reduces the risk of loss of RWCU/SDC events during Mode 5 (from internal events and external events).

GEH responded that loss of RWCU/SDC during shutdown would cause reactor pressure vessel (RPV) pressure and temperature to both increase to near or above Mode 1 conditions. GEH responded that if non-condensable gases accumulate in the IC such that the heat transfer drops below the decay heat of the reactor, the reactor pressure would increase to the setpoint for opening of the ICS vent line. The ICS venting process is the same irrespective of the initial condition operating mode. GEH replied that no specific calculations have been performed for the ICS system with an initial condition of the reactor in Mode 5 since it is bounded by operation from Mode 1 conditions.

The staff does not agree that Mode 1 conditions for successful IC operation bound Mode 5 conditions. The staff requests the following information.

A. *As preparations take place for reactor vessel head removal in Mode 5, the vessel head space will be filled with air. The staff is requesting GEH to document the set point for the automatic opening of the vent valves in the DCD and in Chapter 16 of the TS for IC operability, or GEH should add the operator action to vent the IC in the PRA.*

B. *The staff understands that vessel level will be raised above the level eight (8) setpoint as preparations take place for reactor vessel head removal in Mode 5. The staff is concerned that the IC stub tube could become filled with water which would affect the ability of the IC to function. If GEH intends for the IC to initiate cooling with water in the IC stub tube, please demonstrate the ICS capability to remove decay heat at the highest maximum vessel level before de-tensioning of the reactor vessel head studs is initiated.*

C. *The staff is requesting GEH to document in the DCD the maximum initial vessel levels for IC operability and successful IC operation.*

D. *If there are RCS configurations in Mode 5 operation that will not support IC operability, then the staff is requesting GEH to remove credit of the ICS from the baseline PRA and the RTNSS evaluation for PRA plant operational states, Mode 5 and Mode 5-open.*

GEH Response

A. *ICS venting is explicitly modeled in the fault tree for ICS system. System function is fully dependant on venting. As modeled, if venting fails in any loop, that loop fails in the model. This is true for both the Level 1 model and the Shutdown model. It is more limiting in the Shutdown model since only two ICS loops are available and both are required for success.*

The setpoint for automatic ICS venting of non-condensables is listed in DCD 5.4.6.2.3 (1090 psi for lower header primary vent, lower header bypass vent is 1150 psi). Additionally, the ESBWR PRA has manual venting modeled for the ICS. This action is modeled as a backup to the automatic action though. The basic event in the model is "B32-XHE-FO-VENT – Operator fails to open ICS vent."

B. Vessel level will not be raised during Mode 5. Discussions with engineers developing shutdown and outage planning have clarified this issue. The sequence of events has all of the head bolts being removed prior to elevation of the water level above the dryer. Level will only be raised for shielding/dose concerns prior to head removal, but after all head bolts are removed. This means water level will not be elevated until Mode 6.

C. Providing the maximum vessel level for IC operability to support the Shutdown PRA treatment is unnecessary. Vessel level is not elevated until Mode 6, and ICS is not credited at all during Mode 6.

D. The ESBWR Shutdown PRA credits the ICS in Mode 5 (and Mode 5 Open). This is due to Technical Specifications requiring the system to be available throughout the mode. This is consistent with GEH treatment of other systems in developing the Shutdown PRA.

There is a small disparity between how the modes are defined in the Technical Specification and how they are treated in the Shutdown PRA. Technical Specifications state that Mode 6 is entered when the first head bolt is detentioned. For the Shutdown PRA Mode 6 is defined as having the vessel head no longer providing a pressure seal. This is likely some time after detentioning the first bolt but before vessel head removal.

Due to this small disparity, a sensitivity analysis was conducted. This sensitivity shows the Shutdown PRA results with no ICS credited for a period of Mode 5 Open immediately prior to transition to Mode 6. Cases for the Mode 5 Open without ICS credited are evaluated at 8, 16, and 24 hours. The full details of this analysis will be included in the Revision 4 update to Chapter 16 of NEDO 33201.

The steps in the sensitivity were:

- Edit Mode 5 Open event tree to remove ICS
- Recalculate Initiating Event frequencies for Mode 5 and Mode 5 Open (Mode 5 Open at 8, 16, & 24 hours).

- Quantify Shutdown PRA for each case

	Mode 5 (hours)	Mode 5 Open (hours)	ICS credited in Mode 5 Open	Result
Baseline Shutdown PRA	192	48	Yes	**9.37E-09**
Case - 24HR-M5O	216	24	No	**1.25E-08**
Case - 16HR-M5O	224	16	No	**1.15E-08**
Case - 8HR-M5O	232	8	No	**1.05E-08**

The results show an increase in CDF if ICS is not available for the duration of Mode 5 Open. However, the increase is relatively small and the top cutsets in each case remain the same. The results in all cases are still dominated by lower drywell LOCA events, which are not impacted by ICS availability. Even if a period of Mode 5 does occur with ICS inoperable, it is certain to be less than 24 hours, and will likely be less than 8. The results of the sensitivity show that if an extended period exists during Mode 5 where ICS is inoperable, it does not impact the CDF enough to be classified as a 'key' risk insight.

The full details of the above sensitivity cases will be included in NEDO 33201 Revision 4, Chapter 16.

DCD/NEDO-33201 Impact

No Changes to the DCD will be made in response to this RAI.
NEDO-33201, Rev. 4 will be revised as noted in the above response.

NRC RAI 19.1-144 S03

Question Summary

Provide additional information regarding isolation condenser system operation for cold shutdown conditions. Shutdown IC Operation

Full Text:

The staff reviewed GEH response to RAI 19.1-144 S02. The staff understands that level must be raised at shutdown to remove the reactor vessel head. As documented in RAI 19.1-144S02, the staff is concerned that once the IC stub tube becomes filled with water, the ability of the isolation condenser (IC) to function would be adversely impacted. The staff requested GEH to document in the DCD the highest maximum vessel level at which the ICS can function. The staff also requested GEH to demonstrate the ICS capability to remove decay heat at the highest reactor vessel level before de-tensioning of the reactor vessel head studs is initiated.

GEH then stated in Chapter 22 of the ESBWR PRA, Rev. 3 that "Many current operating plants raise the water level in the vessel prior to vessel head removal during shutdown. This is done to reduce the dose associated with removal of the reactor head and other refueling activities. For the ESBWR, this evolution disables ICS as flooding the vessel above the ICS inlets will disable the system" and "the relatively low risk numbers associated with shutdown are primarily the result of ICS and GDCS". Since IC capability is risk significant, the staff requests the following information:

1. GEH is requested to demonstrate the capability of the ICS to remove decay heat at the documented highest allowed vessel level.

2. GEH is requested to document in the DCD whether opening of the reactor vessel head vent in Mode 5 could impact ICS operation.

3. In GEH RAI response, GEH did not document in the DCD the reactor vessel level at which the ICS becomes non-functional as requested by the staff. GEH only mentioned in the RAI response that vessel level will not be raised during Mode 5. The failure of the operator to maintain proper level for ICS functionality is not modeled in the PRA. The staff considers this potential operator error to be risk-significant. The staff is requesting GEH to document in TS the highest reactor vessel level for ICS operability to ensure that the operators do not raise level such that ICS becomes disabled.

GEH Response

The Shutdown PRA model is being revised in Section 16 of NEDO-33201 Revision 4 to remove credit for ICS during PRA Mode 5-Open, i.e., the period before the RPV head is removed for refueling ("Open" signifies that the containment head is removed). This change will address the issues in this RAI and its supplements. These issues are based on uncertainty about the performance of ICS when the RPV water level is outside of its normally controlled range in PRA Mode 5. This change will only allow credit for ICS in the Shutdown PRA model when the RPV level is controlled at a level that is known to support ICS operation. The effect of this change is a minor increase in Shutdown CDF and LRF, as explained in response to RAI 19.1-144 S02.

RAI 19.1-144 and supplements 1 and 2 requested calculations to demonstrate that ICS could operate following an extended loss of shutdown cooling. GEH responded by stating a long-term loss of shutdown cooling would cause RPV pressure and temperature to rise, and thus ICS would perform in the same manner that it would if there were an at-power RPV isolation. Supplement 3 to this RAI raises the additional concern that ICS may not be functional if RPV water level is raised above the elevation of the ICS stub tubes.

In Technical Specification (TS) Mode 5, the RPV water level is maintained below the ICS stub tubes. TS Mode 6 is entered when the first RPV head bolt is de-tensioned. When the RPV head bolts are removed, and the RPV head is ready to be removed, RPV water level is raised above the top of the steam dryers for shielding purposes. During this time the RPV level rises above the ICS stub tube elevation; however, ICS is already inoperable, and not necessary, because the RPV head has been removed.

The Shutdown PRA considers PRA Mode 6 as when the RPV head no longer provides a pressure seal. This condition occurs after de-tensioning head bolts and before the RPV head is removed. Therefore, there is a small time window in PRA Mode 5-Open (TS Mode 6) when RPV level could be raised to the point where ICS is inoperable. Because operational requirements on RPV level control are not specified until the detailed design phase, it is not known when the RPV water level will rise enough to make ICS inoperable during this evolution. Therefore, it is more practical to remove credit for ICS operation during this time period (PRA Mode 5-Open).

The sensitivity study that was described in the response to RAI 19.1-144 S02 shows that ICS is not risk significant in PRA Mode 5-Open and therefore, the treatment described above is considered to be appropriate.

The specific items requested in this supplement are addressed as follows:

Item 1: Demonstrating the capability of the ICS to remove decay heat at the documented highest allowed vessel level is not necessary to support the Shutdown PRA. This is because, as stated above, operating procedures will maintain vessel level within a controlled band, and the Shutdown PRA is being revised so credit is taken for ICS only in Modes where vessel level is controlled.

Item 2: Opening of the reactor vessel head vent in TS Mode 5 is inconsequential because the average coolant temperature is less than or equal to 200 degrees F. Other postulated scenarios, such as re-entering TS Mode 4 due to loss of decay heat removal represent a departure from normal operations to which the licensed unit's operating staff must be prepared to respond. This preparation is within the scope of emergency procedures that are discussed in DCD Tier 2 Section 13.5, and is not a part of the ESBWR design scope.

Item 3: The sensitivity study provided in GEH response to RAI 19.1-144 Supplement 2 demonstrates that loss of ICS in PRA Mode 5-Open is not risk significant. The ability of operators to maintain proper vessel level such that ICS remains functional is important operational information, however, it does not meet the requirements of 10 CFR 50.36 for inclusion into TS because it is not risk significant.

DCD Impact

No DCD changes will be made in response to this RAI.

LTR NEDO-33201, Rev 4 will be revised as described above.

RAI 19.1-144 S04

The staff has reviewed GEH's response to RAI 19.1-144 S03 and requests the following additional supporting information to resolve the ICS functionality and operability issues during Mode 5 conditions:

a. Provide additional information in the description of ICS in the DCD regarding the ability of the IC stub tube and IC steam line to clear itself as the water level lowers in the vessel.

b. Provide additional information in the PRA concerning the reactor head vent, including the size of the head vent, status of head vent (opened or closed), the discharge path of the head vent, and the duration of time that the head vent can be opened and not impact ICS operation.

c. Provide clarification in Technical Specifications regarding (1) operability of the ICS during reactor vessel high water level (flooded stub tube), and (2) the impact of the Action Statements that allow ICS inoperability for an indefinite period of time.

GEH Response

a. DCD Tier 2 Subsection 5.4.9 is being revised to provide additional information in the description of ICS regarding the ability of the IC stub tube and IC steam line to clear itself as the water level lowers in the vessel.

b. NEDO-33201 Section 16 is being revised to provide additional information concerning the reactor head vent, including the size of the head vent, status of head vent (opened or closed), the discharge path of the head vent, and the duration of time that the head vent can be opened and not impact ICS operation.

c. The GEH response to RAI 16.2-188 MFN 09-357dated June 8 2009)addresses the revision to the ESBWR GTS 3.5.5, Isolation Condenser System (ICS) – Shutdown, and GTS 3.5.5 Bases to provide clarification regarding (1) operability of the ICS during reactor vessel high water level (flooded stub tube), and (2) the impact of the Action Statements that allow ICS inoperability for an indefinite period of time.

DCD Impact

DCD Tier 2 Subsection 5.4.9 is revised as shown in the attached mark-up.

NEDO-33201 Section 16 is revised as shown in the attached mark-up.

ENCLOSURE 2

DCD Revision 6 Markups

Section 5.4.9 Main Steamlines, Steam Stub Lines, And Feedwater Piping
Section 17.4 D-RAP

NEDO 33201 Revision 3 Markups
Section 16.4

RWCU/SDC system requires manual realignment of cross-connections with the FAPCS. Each cross-connection contains spectacle flanges and closed manual isolation valves. These provisions preclude the possibility of intersystem LOCA during normal modes of operation. There is also an intersystem cross-connection, which must be realigned for mid-vessel injection.

The NRHX provides the heat removal capacity to sufficiently cool the plant from stable shutdown conditions to cold shutdown conditions (Table 5.4-3).

5.4.8.2.3 Safety Evaluation

The RWCU/SDC system does not perform or ensure any system level safety-related function, and thus, is classified as nonsafety-related.

Refer to Subsection 5.4.8.1.3 for an evaluation of the safety-related containment isolation, and instrumentation for pipe break detection outside the containment functions of the RWCU/SDC system.

Loss of RWCU/SDC function due to vessel level decrease below the first stage spillover level would result in heatup and expansion of the vessel coolant inventory inside the shroud. A minor decrease in level due to thermal contraction from cooldown is, therefore, self correcting. Loss of decay heat removal due to a more significant decrease in vessel level is bounded by the evaluation of a total loss of RWCU/SDC function provided in Subsection 15.2.2.9, and by the evaluation of the spectrum of postulated LOCA events described in Section 6.3.

5.4.8.2.4 Testing and Inspection Requirements

Refer to Subsection 5.4.8.1.4 for the testing and inspection requirements for the RWCU/SDC system.

5.4.8.2.5 Instrumentation

Each pump is protected from potential cavitation during the shutdown cooling mode by a speed runback set to actuate if the RPV water level falls to Level 3. RWCU/SDC system instrumentation is described in Subsection 7.4.3. The shutdown cooling mode of the RWCU/SDC has an automatic temperature control function that controls the speed of the ASDs to control the coolant temperature as measured by the core inlet thermocouples during the shutdown operation.

Instruments monitoring the temperature of the RCCWS water leaving the NRHX also automatically control the RWCU/SDC system flow by adjusting the pump speed in the event the RCCWS outlet temperature from the NRHX rises above limit.

5.4.9 Main Steamlines, Steam Stub Lines, and Feedwater Piping

5.4.9.1 Design Bases

Safety Design Bases

The main steam and feedwater lines are designed to:

- Withstand the stresses from internal pressures, safe shutdown earthquake (SSE) loads, DBA loads, hydrodynamic loadings, reactions from discharging SRVs and SVs (for the main steamlines) or DPVs (for the steam stub lines), for ICS initiation and operation (for the steam stub lines), loads from fast closure of the turbine stop and/or control valves (for the main steamlines), and waterhammer loads (for the feedwater lines); and

- Provide for long-term leak-tight isolation of the RPV and the containment.

Power Generation Design Bases

The main steam and feedwater lines are designed to:

- Transport steam from the reactor vessel through the steamlines over the full range of reactor power operation and, in conjunction with the MSIVs, limit the pressure drop from the reactor to the turbine to less than the design value;
- Supply water to the reactor vessel through the feedwater lines over the full range of reactor power operation; and
- Permit flooding of the steamlines up to the main turbine stop valves during refueling and other shutdowns without the need for adding temporary supports.

5.4.9.2 Description

The main steamlines consist of carbon steel piping originating at reactor vessel nozzles and running to the main steamline header in the turbine building. From the main steamline header, there are four lines that run to and terminate at the turbine stop valves. The feedwater lines are low alloy steel piping beginning from the interface at the seismic restraint just inside the steam tunnel through containment penetration into the drywell and then branching to lines connecting to reactor vessel nozzles. The main steam and feedwater piping from the reactor through the isolation valves in the reactor building is shown schematically in Figure 5.4-3. Further descriptions of the main steamlines downstream of the outboard MSIVs and the feedwater lines upstream of the seismic restraint for the outboard isolation valves are contained in Sections 10.3 and 10.4, respectively.

The main steamlines are Quality Group A and ASME Section III, Class 1 from the RPV through the outboard MSIVs. They are Seismic Category I from the RPV to the seismic interface restraint downstream of the outboard MSIV. The main steamlines from the outboard MSIV to the turbine stop valves are described in Section 10.3 and Table 3.2-1.

The four steam stub lines consist of low alloy steel piping originating at the reactor vessel nozzles and running to the respective ICS train steam supply line interface connection, and include pairs of DPVs mounted at the terminal ends. The DPVs are described in Subsection 5.4.13. The steam stub lines are Quality Group A, ASME Section III, Class 1, and Seismic Category I. The steam stub lines are mounted to the RPV as nominally horizontal piping, sloped back to the reactor vessel to assure moisture drainage away from the ICS steam line or the DPV inlets.

The feedwater lines are Quality Group A and ASME Section III, Class 1 from the RPV through the outboard containment isolation valves, including the branch isolation valves; Quality Group B and ASME Section III, Class 2 from the outboard containment isolation valves to the seismic interface restraints. They are Seismic Category I from the RPV to the seismic interface restraint and Seismic Category II from the seismic interface restraint to the last feedwater heater.

17.4 RELIABILITY ASSURANCE PROGRAM DURING DESIGN PHASE

This section presents the ESBWR Design Reliability Assurance Program (D-RAP).

17.4.1 Introduction

The GEH ESBWR D-RAP is a program utilized during detailed design and specific equipment selection phases to assure that the important ESBWR reliability assumptions of the Probabilistic Risk Assessment (PRA) are considered throughout the plant life. The PRA is used to evaluate the plant response to initiating events and mitigation to ensure potential plant damage scenarios pose a very low risk to the public.

The D-RAP identifies relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for owner/operator consideration in assuring safety of the equipment and limiting risk to the public. An example is provided in Subsection 17.4.11 to demonstrate how the D-RAP applies to the Isolation Condenser System (ICS). The ICS example shows how the principles of D-RAP are applied to other systems identified by the PRA as being risk-significant.

The COL Applicant will identify the site-specific SSCs within the scope of the Reliability Assurance Program (RAP), and describe the quality elements for developing and implementing the D-RAP (that is, Organization, Design Control, Procedures and Instructions, Records, Corrective Action, and Audit Plans) that will be applied prior to the initial fuel load (COL 17.4-1-A).

The COL Applicant~~Holder~~ will provide a description of operational reliability assurance activities (COL 17.4-2~~1~~-A~~H~~). These activities are consistent with the following requirements:

- Integrate the objectives of operational reliability assurance activities into the QA program, including addressing failures of nonsafety-related, risk-significant SSCs that result from design and operational errors in accordance with SECY 95-132, Item E.

- Establish PRA importance measures, the expert panel process, and deterministic methods to determine the site-specific list of risk-significant SSCs under the scope of the D-RAP.

- Evaluate and maintain the reliability of risk-significant SSCs as identified in the D-RAP. This includes determining the dominant failure modes of risk-significant SSCs. The program may cite, for example, reliability analysis, cost-effective maintenance enhancements, such as condition monitoring and using condition-directed maintenance as well as time directed or planned periodic maintenance.

- Use the Maintenance Rule (10 CFR 50.65) program to monitor the effectiveness of maintenance activities needed for operational reliability assurance.

- Consider all SSCs that are in the scope of the D-RAP as high-safety-significant (HSS) within the scope of the Maintenance Rule program, or provide Expert Panel justification for any exceptions.

  Note: The Expert Panel, in accordance with common industry practice and guidance in NUMARC 93-01, develops the final list of risk significant SSCs from various inputs, including the PRA risk importance calculations and industry operating experience. It is necessary for the Expert Panel to include all SSCs that are in the scope of the RAP to be

17.4.13 COL Information

17.4-1-A Identifying Site-Specific SSCs Within the Scope of the RAP

The COL Applicant will identify the site-specific SSCs within the scope of the RAP, and describe the quality elements for developing and implementing the D-RAP (that is, Organization, Design Control, Procedures and Instructions, Records, Corrective Action, and Audit Plans) that will be applied prior to the initial fuel load (Subsection 17.4.1).

17.4-~~1~~2-~~H~~A Operation Reliability Assurance Activities

The COL Applicant~~Holder~~ will provide a description of operational reliability assurance activities (Subsection 17.4.1).

17.4.14 References

17.4-1 GE Energy Nuclear, "Reliability Assurance Program Plan", NEDO-33289, Revision 1, December 2007.

17.4-2 US Nuclear Regulatory Commission, "Policy and Technical Issues Associated With the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)", SECY-95-132, May 1995.

17.4-3 US Nuclear Regulatory Commission, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Regulatory Guide 1.160, March 1997.

17.4-4 US Nuclear Regulatory Commission, "Assessing and Managing the Risk Before Maintenance at Nuclear Power Plants," Regulatory Guide 1.182, May 2000.

17.4-5 NEI, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," NUMARC 93-01 April 1996.

17.4-6 NEI, "Industry Guidance for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision to Section 11 of NUMARC 93-01 February 22, 2000.

17.4-7 GE Hitachi Nuclear Energy, "Risk Significance of Structures, Systems and Components for the Design Phase of the ESBWR," NEDO-33411, Revision 0, March 2008.

17.4-8 GE Hitachi Nuclear Energy, "Quality Assurance Program," NEDO-33181, Revision 5, February 2008.

The IC function is able to prevent RPV Level 1 from being reached if:

- The initial RPV water inventory is above Level 3
- There is little or no leakage from the RPV.

The maximum RCPB leak rate within the Technical Specification during full power operation is assumed to be insufficient to decrease the level to the point where an ADS signal occurs, even if high pressure RPV makeup is not established during the sequence mission time. Therefore, failure of the IC function due to leaks is considered a low probability.

The success criterion of this function is the operation of both operable (2/2) ICs during the sequence mission time. The Tech Specs for Mode 5 only require 2 out of the 4 ICs be available.

In Tech Spec Mode 5, the RPV water level is normally maintained below the ICS stub tubes. Per DCD Tier 2, Subsection 5.4.9.2, the four steam stub lines consist of low alloy steel piping originating at the reactor vessel nozzles and running to the respective ICS train steam supply line interface connection, and include pairs of DPVs mounted at the terminal ends. The steam stub lines are mounted to the RPV as nominally horizontal piping, sloped back to the reactor vessel to assure moisture drainage away from the ICS steam line or the DPV inlets. Therefore, even under a postulated scenario that the ICS stub tubes are flooded originally, the loss of DHR events would eventually result in the boiling of the coolant, the RPV water level would drop below the ICS stub tubes' elevation and ICS can perform its DHR function, which then is self-correcting.

Per DCD Tier2, Subsection 5.4.12, the ESBWR has an RPV head vent system that handles any noncondensable gas buildup, that could inhibit natural circulation core cooling, at the high point inside the RPV head by sweeping the gasses through a main steamline and then ultimately to the condenser. The piping is two inches in diameter. The vent piping directs air and non-condensable gases from the RPV to either the Equipment and Floor Drain Sump or one of the main steamlines. Per DCD Tier 2, Subsection 5.4.12.2, the RPV head vent remains open to the MSLs during normal power operation and following any postulated transient or accident. The motor-operated shutoff valve is designed to remain open, and is not required to perform an active safety function. The alternate path vent line to the equipment and floor drain sump system is normally closed to protect the RCPB and the nitrogen-operated isolation valves for this line are designed to remain shut following a postulated transient or accident. During reactor shutdown and after the plant reaches cold shutdown conditions, the two valves in the vent piping leading to the Equipment and Floor Drain Sump are opened and the valve in the piping connected to the main steamline is closed.

The duration of time the head vent can be open in Mode 5 is inconsequential because the average coolant temperature is below or equal to 200°F. In a postulated unplanned re-entry into Mode 4 from Mode 5 due to loss of DHR, the head vent should not impact the ICS operability because the isolation of this line should be considered to be very likely. Moreover, if such opening is assumed to be a shutdown LOCA event, it has already been bounded by the existing event trees for LOCA other than feedwater or GDCS, which have negligible risk contributions.

MS-TOP18~~-~~: At Least 1 SRV Open

This is the same heading as is used for the full power model.