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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 132 Related to ESBWR Design Certification Application, RAI Numbers 19.1-156 through 19.1-159, 19.1-165 through 19.1-170 and 22.5-20**

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) sent by NRC letter dated January 15, 2008 (Reference 1). The GEH responses to RAI Numbers 19.1-156 through 19.1-159, 19.1-165 through 19.1-170 and 22.5-20 are in Enclosure 1.

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

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN-08-040. Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, *Request For Additional Information Letter No. 132 Related To ESBWR Design Certification Application,* January 15, 2008

Enclosures:

1. Response to Portion of NRC Request for Additional Information Letter No. 132 Related to ESBWR Design Certification Application, ESBWR Probabilistic Risk Assessment, RAI Numbers 19.1-156 through 19.1-159, 19.1-165 through 19.1-170 and 22.5-20

2. Enclosure 1, Response to Portion of NRC Request for Additional Information Letter No. 132 Related to ESBWR Design Certification Application ESBWR Probabilistic Risk Assessment RAI Numbers 19.1-156 through 19.1-159, 19.1-165 through 19.1-170 and 22.5-20

3. Attachment 1, DCD Markup of Section 19.4.2 PRA Maintenance and Update Program (RAI 19.1-156); DCD Markup of Section 19.2 PRA RESULTS AND INSIGHTS (RAI 19.1-157); DCD Markup of Section 19.2.4.1.3 Flooding During Shutdown (RAI 19.1-158); DCD Markup of Section 19.2.4.3 Significant Offsite Consequences of Shutdown Mode (RAI 19.1-159); DCD Markup of Section 19.A.3.2 Seismic Assessment (RAI 22.5-20)

cc:	AE Cubbage	USNRC (with enclosure)	
	GB Stramback	GEH/San Jose (with enclosure)	
	RE Brown	GEH/Wilmington (with enclosure)	
	DH Hinds	GEH/Wilmington (with enclosure)	
	eDRF Sections	0000-0081-9166	NRC RAI 19.1-156
		0000-0081-9174	NRC RAI 19.1-157
		0000-0081-9178	NRC RAI 19.1-158
		0000-0081-9183	NRC RAI 19.1-159
		0000-0081-2079	NRC RAI 19.1-165
		0000-0081-2087	NRC RAI 19.1-166
		0000-0081-2088	NRC RAI 19.1-167
		0000-0081-2092	NRC RAI 19.1-168
		0000-0081-2093	NRC RAI 19.1-169
		0000-0081-8303	NRC RAI 19.1-170
		0000-0081-9195	NRC RAI 22.5-20

Enclosure 1

MFN 08-199

Response to Portion of NRC Request for

Additional Information Letter No. 132

Related to ESBWR Design Certification Application

ESBWR probabilistic Risk Assessment

RAI Numbers 19.1-156 through 19.1-159,

19.1-165 through 19.1-170 and 22.5-20

NRC RAI 19.1-156

The Staff is requesting GEH to update DCD, Tier 2, Revision 4, Section 19.4.2 PRA Maintenance and Update Program as follows, "The PRA will be updated to reflect plant design, operational, and PRA modeling changes, consistent with NRC-endorsed standards in existence 1 year prior to issuance of the update, which will be prior to initial fuel load, and then every four years. The key assumptions in the PRA as documented in DCD Tier 2, Table 19.2-3 will be maintained or any departures shall be addressed. The COL Holder maintains this information in accordance with documentation and records retention requirements."

GEH Response

DCD Tier 2, Revision 5, Section 19.4.2 will be revised to add the requested information.

DCD Impact

DCD Tier 2, Revision 5 will be revised in accordance with the attached draft markup.

NRC RAI 19.1-157

The staff is requesting GEH to clarify the definitions of the disposition of key risk assumptions and risk insights listed in Column 2 in DCD, Tier 2, Revision 4, Table 19.2-3, "Risk Insights and Assumptions" as follows:

A. GEH should clarify that "Operational Program" means that development of Operating and Maintenance Procedures is the responsibility of the COL Applicant in accordance with COL Item 13.5-2-A. These PRA assumptions will be addressed by the COL applicant as part of this COL action item.

B. GEH should provide reference to a specific DCD section for each item identified as a "design requirement."

GEH Response

A. DCD, Tier 2, Section 19.2.1 will be revised to clarify that "Operational Program" means that development of operating and maintenance procedures is the responsibility of the COL Applicant in accordance with COL Item 13.5-2-A.

B. GEH will provide a specific DCD section for each item identified as a design requirement in DCD Revision 5.

DCD Impact

DCD Section 19.2.1 will be revised in response to item "A" as shown on the attached markup.

DCD Section 19 will be revised in response to item "B" as described in the GEH response.

NRC RAI 19.1-158

It is stated in Section 19.2.4.1.3 of the DCD, Revision 4, that the shutdown flooding CDF is negligible. Based on comparison of the reported shutdown internal events CDF and shutdown flooding CDF, GEH is requested to justify or revise this statement in the DCD.

GEH Response

Section 19.2.4.1.3 of DCD, Revision 4 will be revised to delete the statement that shutdown flooding CDF is negligible and to clarify its relative importance. The conclusion for flooding during shutdown does not change, that is, the CDF is a very low number and there are no significant insights.

DCD Impact

Section 19.2.4.1.3 of DCD, Revision 4 will be revised as shown on the attached markup.

NRC RAI 19.1-159

In Section 19.2.4.2 of the DCD, Revision 4, GEH states that, "The offsite consequences from shutdown risk are judged to be negligible on the following basis: "The significant shutdown events occur during Mode 6, which does not begin until approximately 96 hours after shutdown. The decay of fission products after 96 hours reduces the source term to less than 1% of the value at power operating conditions." The staff is requesting GEH to justify or revise this statement based on the following: As reported in Chapter 16 of the PRA, over 40% of the internal shutdown CDF is related to Mode 5. In addition, in NUREG/CR 6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events", on pages 4-3, it states, "The results indicate that source terms which involve a release of about 10% or less of the core iodine inventory (10% iodine releases are associated with early fatalities in accidents that occur at full-power), offsite doses generally fall below the early fatality threshold approximately 8 days or less after shutdown." Therefore, the consequences of severe accidents occurring during Modes 5 and 6 approximately 8 days or less after shutdown should not be characterized as negligible in the DCD, Chapter 19.

GEH Response

Section 19.2.4.3 of the DCD, Revision 4 will be revised to delete the statements regarding: "significant shutdown events occurring during Mode 6"; and "the decay of fission products after 96 hours reduces the source term to less than 1% of the value at power operating conditions." The source terms for containment bypass events may not fall below the early fatality threshold until approximately 8 days after shutdown; however, the frequency of shutdown containment bypass events is very low. As a result the offsite consequences, which are the product of the source term risk and the shutdown containment bypass frequency, are not significant.
If the radiological release inventories from the at power analysis in NEDO 33201 Revision 2, Table10.4-1b "MACCS2 Results by Source Term 72 Hour After Onset of Core Damage" are used as an upper bound estimate of the release inventories that would occur after considerable decay time during a shutdown event, and the CDF values for Modes 5 and 6 from NEDO 33201 Revision 2, Table 16.6-1, "Shutdown CDF by Initiating Event and by Mode of Operation" are used to estimate the large release frequency, (assuming 100% of core damage events in Modes 5 and 6 result in containment bypass), then combining the large release frequency with the release fractions yields the offsite consequences. They are calculated to be less than 1% of the target values listed in NEDO 33201 Revision 2, Table 10.4-2, "Baseline Consequence Goals and Results." Therefore, the consequences of severe accidents during Modes 5 and 6 are not significant.

DCD Impact

Section 19.2.4.3 of the DCD, Revision 4 will be revised per the attached markup.

RAI Response 19.1-165

Please describe what risk assessment sensitivity studies were performed for hurricanes in the high winds risk assessment, and what the results and insights were. If sensitivity studies were not performed, please explain the basis for not performing any?

GEH Response

No sensitivity studies were conducted for the hurricane high winds risk assessment. However, a bounding analysis approach is applied to the calculation of the ESBWR hurricane high winds core damage risk. Data representative of ONLY coastal regions are used to generate the initiating event frequency of 1.52 E-02 events/rcy for hurricane related losses of off-site power. In contrast, a smaller initiating event frequency of 6.01E-03 events/rcy represents the hurricane related losses of offsite power of ALL plants.

By applying the more conservative data associated with the coastal region, a bounding CDF for the hurricane contribution to the high winds risk assessment is obtained and represents the ESBWR hurricane high winds risk analysis. The following table illustrates the bounding analysis approach in the ESBWR hurricane high winds risk assessment.

Parameter	Hurricane High Winds Risk Analysis	
	ESBWR	All US Plants (including non-coastal)
Initiating Event Frequency - Hurricane	1.52E-02	6.01E-03
CDF- At-power Hurricane	1.29E-09	5.07E-10
CDF - At-power High Wind	1.34E-09	5.56E-10

DCD/NEDO-33201 Impact

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

No change to NEDO-33201, Rev. 3 will be included in response to this RAI.

RAI Response 19.1-166

Please explain whether there are above-ground, outdoor tanks or other structures holding significant quantities of liquids, such as water or oil that if failed or damaged could cause a flooding issue for other important equipment on site (e.g., pumps, transformers)? If so, please describe the tank(s) including whether they are protected by Seismic Category 1, or Category 2 structures, and any flooding mitigative features on the ESBWR site that would reduce the probability or consequences of undesirable events.

GEH Response

The ESBWR flooding risk analysis considers potential sources of flooding including both indoor and outdoor tanks and other structures containing liquids. These tanks are evaluated to determine their maximum liquid capacity. In addition, the risk analysis estimates surface area of the buildings/structures containing these tanks, and identified PRA components located within these areas.

Using this information, screening is conducted to eliminate tanks/structures from the flood risk analysis that present minimal impact as a potential flooding source. Factors for screening tanks include the following:

- No PRA components in building/structure, and
- Insufficient volume as defined by a flood of available surfaces of no greater than 1-foot deep.

For the ESBWR flooding risk analysis, tanks or other structures located outside in the yard are screened based on both of the aforementioned screening criteria. As a result, no above-ground, outdoor tanks are identified as potential flooding sources.

DCD/NEDO-33201 Impact

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

No change to the NEDO-33201, Rev. 3 will be made in response to this RAI.

RAI Response 19.1-167

Please describe specifically how the effects of tornado missiles on Seismic Category II and RTNSS structures were accounted for in the ESBWR high winds PRA. If such effects were not modeled, explain the reason(s)/justification for not performing the evaluation.

GEH Response

Section 14.9 reference 14.9-2 provides the following bases for the calculation of the high winds risk assessment:

- "ESBWR Standard Plant structures, which are Seismic Category I, are designed for tornado and extreme wind phenomena. Seismic Category II structures are designed for extreme and tornado wind (excluding tornado missiles)."
- Wind speed design parameters for RTNSS "...hurricane wind speed (3-second gust) shall be taken as 87.2 m/sec (195 mph)..."
- In addition, standard missile impact for RTNSS is designed for "hurricane wind speed (87.2 m/sec (195 mph) 3-second gust)..."

In conducting the ESBWR high winds risk analysis, the seismic design criteria and location of system components is identified. Components designed for Seismic Category I or located with buildings that are designed as Seismic Category I structures are assumed to perform their function during all high wind scenarios, including tornado missiles. Components designed as Seismic Category II or RTNSS structures that are not housed in Seismic Category I buildings are considered to be susceptible to tornado missiles and no credit is taken for the function of these components during the F4/5 tornado high wind analysis.

High Wind Risk Analysis	Hurricane Category	Wind Speed[1] (mph)		ESBWR Plant Structures[4]			
				SC-I	SC-II	RTNSS	NS
Hurricane	Cat. 3/4/5	> 155		✔	✔	✔	x
High Wind Risk Analysis	Tornado Category	Wind Speed 3-sec gust (mph)		ESBWR Plant Structures			
		Fujita[2]	EF Scale[3]	SC-I	SC-II	RTNSS	NS
F2/3	F2/EF2/F3/EF3	118 - 161	110 - 137	✔	✔	✔	x
F4/5	F4/EF4/F5/EF5	262 - 317	200 - 234	✔	x	x	x

Notes:

1 Based on Saffir-Simpson Hurricane Scale

2 Based on Fujita Scale

3 Based on Enhanced Fujita Scale

4 "✔" indicate no damage to structure sustained; "x" indicate structure will sustain damage.

DCD/NEDO-33201 Impact

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

No change to the NEDO-33201, Rev. 3 will be made in response to this RAI.

RAI Response 19.1-168

In discussions with GEH, it was indicated that the ESBWR high winds PRA does not model damage to unprotected equipment in the area surrounding plant structures (e.g., fire hydrants) during hurricane and tornado events. GEH also stated that some credit is taken in the high winds PRA for use of the fire water system to cool the core or refill water tanks at the top of containment in hurricane and tornado loss of preferred power events. Please provide the effect on CDF, as well as risk insights, if no credit is given for the use of the fire water system (in whole or in part) to cool the core or refill tanks following a tornado or hurricane event?

GEH Response

Within the ESBWR PRA, functions of the fire protection system (FPS) system that are credited in the PRA include (1) makeup water to the ICS/PCCS pool, and (2) reactor water coolant/inventory control. To support these PRA functions, the following FPS components are required and designed as seismic category I and seismic category II:

- Primary fire water storage tanks (SC-I)
- Fire pump enclosure (SC-I)
- Primary diesel-driven fire pump (SC-I)
- Primary motor-driven fire pump (SC-II)
- Primary diesel fire pump fuel tank (SC-I), and
- Piping and valves including supports (including source of makeup water to IC/PCC and fuel pools) (SC-I).

The primary fire water storage tanks are designed as seismic category I and function independently of the fire suppression function (i.e., yard hydrant and piping). Based on the ESBWR FPS design, failures of the fire suppression functions do not impact the success of the FPS in providing makeup water to ICS/PCCS or as reactor water. As such, FPS components that do not support or impact the PRA functions are not included in the high winds risk analysis.

Within the ESBWR PRA, FPS equipment designed as seismic category I is credited as functioning during all tornado and extreme wind scenarios. Seismic category II components are assumed to fail during tornado F4/5 high wind events.

The PRA high winds analysis addresses component failures of the FPS during high wind events appropriately. The at power high winds CDF of 1.34E-09 accurately reflects the FPS high wind event failures.

DCD/NEDO-33201 Impact

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

No change to the NEDO-33201, Rev. 3 will be made in response to this RAI.

RAI Response 19.1-169

Please explain the basis for assuming in the high winds risk assessment that no hurricane or tornado will significantly damage any ESBWR Seismic Category 1 and 2 structure.

GEH Response

Please refer to GEH response to RAI 19.1-167 for a discussion and table providing information on wind speeds and impact to ESBWR structures.

DCD/NEDO-33201 Impact

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

No change to the NEDO-33201, Rev. 3 will be made in response to this RAI.

RAI Response 19.1-170

If it is assumed that the tornado and hurricane risk assessments are bounding for all or most ESBWR sites, please explain the manner in which the assessments bound site-specific assessments that would be associated with plants sited on the coast of the U.S. or in the central or south-east portions of the U.S.

GEH Response

A discussion of the methodology in providing a bounding hurricane high wind risk analysis for the ESBWR is provided in the GEH response to RAI 19.1-165.

Tornado frequencies are generated based on the national average for US sites. To address any uncertainties associated with the ESBWR tornado risk analysis, the tornado frequency is increased by a factor of 10. By using a higher value for the tornado frequency, a bounding CDF for the F2/F3 and F4/F5 tornado high wind risk assessment is obtained and represents the ESBWR high wind risk assessment.

DCD/NEDO-33201 Impact

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

No changes to NEDO-33201, Rev. 3 will be made in response to this RAI.

NRC RAI 22.5-20

In DCD, Revision 4, Section 19A.3.2, GEH stated that the seismic margins analysis is described in DCD, Section 19.2.3.5. This reference appears to be incorrect since the DCD does not include such section. The staff requests GEH to confirm that the correct reference to the description of the seismic margins analysis should be to DCD Section 19.2.3.2.4 or to provide the correct reference. Also, the staff requests GEH to update the DCD as necessary.

GEH Response

The correct reference to the description of the seismic margins analysis is DCD Section 19.2.3.2.4.

DCD Impact

DCD Section 19A.3.2 Revision 5 will be updated to state the correct section. As shown on the attached markup.

MFN 08-199

Attachment 1

DCD Markup of Section 19.4.2
PRA Maintenance and Update Program
(RAI 19.1-156)

DCD Markup of Section 19.2
PRA RESULTS AND INSIGHTS
(RAI 19.1-157)

DCD Markup of Section 19.2.4.1.3
Flooding During Shutdown
(RAI 19.1-158)

DCD Markup of Section 19.2.4.3 Significant Offsite
Consequences of Shutdown Mode
(RAI 19.1-159)

DCD Markup of Section 19.A.3.2 Seismic Assessment
(RAI 22.5-20)

19.4 PRA MAINTENANCE

19.4.1 PRA Design Controls

PRA design controls consistent with the regulatory positions in Regulatory Guide 1.200 contain the following elements:

Personnel performing PRA analyses possess sufficient expertise based on training and job experience to perform the tasks.

Personnel performing technical reviews and independent verifications of PRA analyses possess sufficient expertise based on training and job experience to perform the tasks.

Procedures are in place that control documentation, including revisions to controlled documents and maintenance of records.

Procedures are in place that provide for independent verifications of calculations and information used in the PRA.

Procedures are in place that address corrective actions if assumptions, analyses, or information used previously are changed or are found to be in error

19.4.2 PRA Maintenance and Update Program

The PRA model is a controlled document containing the detailed information for the model. In order to maintain a PRA model that reasonably reflects the as-built and as-operated characteristics of the plant, administrative controls are implemented to:

- Monitor PRA inputs and collect new information;
- Maintain and upgrade the PRA model to be consistent with the as-built and as-operated plant;
- Ensure that cumulative impacts of pending changes are considered in PRA applications;
- Evaluate the impact of PRA changes on previously implemented risk-informed applications;
- Maintain configuration control of the computational methods used to support the PRA model; and
- Document the PRA model and the procedures which implement these controls.

The update process addresses those activities associated with maintaining and upgrading the PRA model and documentation. PRA updates include a general review of the entire PRA model, incorporation of recent plant data and physical plant changes, conversion to new software versions, implementation of new modeling techniques as appropriate, and documentation that facilitates review of PRA changes.

When reviewing pending changes, the impact on the CDF and LRF are estimated. As a result of the estimate, one of the following should occur:

- If the effect of the change is risk significant, a PRA model update is implemented promptly (commensurate with the safety significance of the pending change) without waiting for the normal update cycle.
- If the effect of the change is small the incorporation of the change occurs in the next scheduled model update. The identified change is documented in a change control process.
- If the change has no effect, then no further action is required.

The PRA will be updated to reflect plant design, operational, and PRA modeling changes, consistent with NRC-endorsed standards in existence 1 year prior to issuance of the update, which will be prior to initial fuel load, and then every four years. The key assumptions in the PRA as documented in DCD Tier 2, Table 19.2-3 will be maintained or any departures shall be addressed. The COL Holder maintains this information in accordance with documentation and records retention requirements.

PRA updates are generally consistent with the positions established in Section 1.4 of Regulatory Guide 1.200.

Plant specific design, procedure, and operational changes are reviewed for risk impact. Additional reviews to identify information which could impact the PRA models are completed, including comparison of the PRA model with the knowledge of industry and plant experiences, information, and data with the purpose of identifying inputs pertinent to the PRA. This PRA information includes modeling errors discovered during routine use of the PRA or new information that could impact PRA modeling assumptions.

Various information sources are monitored on an ongoing basis to determine changes or new information that affect the model, model assumptions, or quantification. Information sources include operating experience, technical specification changes, plant modifications, maintenance rule changes, engineering calculation revisions, procedure changes, industry studies, and NRC information.

Once the PRA model elements requiring change are identified, the PRA computer models are modified and appropriate documents revised. Documentation of modifications to the PRA model include a comparison of the prior and the updated results portions delineating the significant changes in the PRA model elements with an associated explanation. The comparison of results provides reasonable assurance that the model update reflects the as-built and as-operated plant.

An independent review of the model or model elements by a qualified reviewer or reviewers is required as part of the update process. When major methodology changes or upgrades are made during an update, the PRA is reviewed by outside PRA experts such as Industry peer review teams and the comments incorporated to maintain the PRA current with industry practices. Peer review findings are entered into the configuration controls process. PRA upgrades receive a peer review for those elements of the PRA that are upgraded.

PRA models and applications are documented in a manner that facilitates peer review as well as future updates and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases. PRA documentation is developed such that traceability and reproducibility is maintained.

19.2 PRA RESULTS AND INSIGHTS

19.2.1 Introduction

This section provides an overview of the ESBWR PRA and a summary of the PRA results. The overview includes the internal and external events analyses, the shutdown PRA, the severe accident progression analysis and the offsite consequence analysis. The ESBWR PRA (Reference 19.1-1) is a full scope (Level 1, 2, and 3) PRA, that covers both internal and external events, for at-power and shutdown operations. Where applicable, ASME-RA-Sb-2005 (References 19.2-2 thru 19.2-4) capability category 2 attributes are included in the analysis. Obviously, some of these attributes are not achievable at the design certification stage of a nuclear power plant. For example, many aspects of assessing human actions cannot be analyzed in absence of a physical, operating plant and operation staff. In these cases, a bounding approach is taken to encompass all potential sites, configurations, and operating organizations. In addition, any analyses requiring site-specific characteristics that are not yet available are treated in a bounding manner.

In cases where detailed design information is not available, or when it can be shown that detailed modeling does not provide additional risk-significant information, bounding assumptions are made. Table 19.2-3 is a list of significant PRA insights and assumptions regarding how the design features affect the risk profile, and how uncertainties affect the PRA model in representing an estimate of the risks of the plant. A systematic method is used to identify PRA insights and assumptions, and to distinguish those that could have a significant effect on the PRA results if alternative assumptions were used. In order to ensure that this information is incorporated into the design process, the PRA insights and assumptions are categorized as follows:

Design Requirement: an assumption that requires specific design details be preserved to maintain its validity.

Operational Program: an assumption that requires specific operational procedures or training be preserved to maintain its validity. Development of operating and maintenance procedures is the responsibility of the COL Applicant in accordance with COL Item 13.5-2-A.

Insight: an assumption that provides significant information about the PRA model or its results that should be maintained in PRA model development, updates, and should be considered when developing conclusions regarding risk-informed decisions.

In order to maintain a PRA model that reasonably reflects the as-built and as-operated characteristics of the plant, controls are implemented to maintain the PRA, as described in Section 19.4.

19.2.2 Uses of PRA

19.2.2.1 Design Phase

The PRA supports the design through assessing risks using key parameters such as Core Damage Frequency (CDF), Large Release Frequency, and importance measures such as Fussell-Vesely (F-V) and Risk Achievement Worth (RAW) for major component functions. In particular, the

Th~~he~~ accident sequences involve line breaks below the top of active fuel, with failure to close the lower drywell equipment hatch (which is assumed to be open during Mode 6), and subsequent failure to flood containment to above top of active fuel. The fourth sequence involves loss of preferred power, with failure to align fire protection system water for injection to the RPV.

The most important operator action in the ESBWR shutdown analysis is to close the lower drywell hatches upon the detection of a break in the RCS. Other operator actions are non-significant contributors to internal events shutdown CDF.

Random failures of individual SSCs are not significant contributors to internal events shutdown CDF.

19.2.4.1.2 Fire During Shutdown

Important fire initiating events in the shutdown internal fires PRA are fires in the Turbine Building that cause a loss of RWCU Shutdown Cooling, and fires in the Service Water structure that cause a loss of Service Water. Failure of the corresponding safety system division is assumed, along with failure of one train of RWCU/SDC and CRD, depending on the particular zone that contains the fire.

The important operator actions in the shutdown internal fires PRA are failure to use CRD injection and failure to use the diesel driven makeup pump for low pressure injection.

19.2.4.1.3 Flooding During Shutdown

The important flood initiating events in the shutdown internal flooding PRA are a failure of a GDCS pool during Mode 6-Unflooded and a CRD break in the Reactor Building during Mode 6. However, the total CDF contribution due to flooding during shutdown sequences is ~~negligible~~ not significant.

19.2.4.1.4 High Winds During Shutdown

Similar to the full power risk profile, the shutdown risk for high winds are limited to Loss of Preferred Power events with a potential loss of the Condensate Storage Tank.

Operator actions are non-significant contributors to the shutdown high wind risk profile. Random failures of systems, structures or components are not significant contributors to the internal events shutdown CDF.

19.2.4.1.5 Seismic Events During Shutdown

Similar to the full power risk profile, the shutdown risk for high winds are limited to Loss of Preferred Power events with a potential loss of the Condensate Storage Tank.

Operator actions are non-significant contributors to the shutdown high wind risk profile. Random failures of systems, structures or components are not significant contributors to the internal events shutdown CDF.

19.2.4.1.6 Shutdown PRA Assumptions

Compared to Residual Heat Removal System in BWRs, the RWCU/SDCS in the ESBWR does not have the potential for diverting RPV inventory to the suppression pool through the SP suction, return, or spray lines.

The arrangement for preventing vessel draining through the design of the control rod drive mechanism (CRDM) is the same as the one used in the ABWR. Therefore, the ESBWR design does not introduce a new challenge to vessel inventory relative to CRDMs.

It is assumed that both RWCU/SDCS trains are running, because the time periods in which only one is running occurs when the reactor well is flooded. Consequently, failure of one of the trains is not considered an initiating event.

Any break above level L3 does not constitute an initiating event, as RWCU/SDC will continue to ensure normal core cooling.

19.2.4.2 Significant Large Release Sequences of Shutdown Mode

Because the majority of the shutdown CDF occurs during times when the containment is open, shutdown modes are not analyzed for large release frequency. Shutdown core damage events can be conservatively assumed to be large releases.

19.2.4.3 Significant Offsite Consequences of Shutdown Mode

~~The dominant contributors to shutdown CDF involve sequences during Mode 6 (Refueling Mode). The dominant initiating events are line breaks from lines penetrating the reactor vessel below the top of the core. In the line break sequences, the critical action is to isolate the lower drywell, by closing the lower drywell hatches, so a boundary can be established to permit flooding above the top of active fuel. The resultant release during a severe accident is considered a containment bypass release.~~

~~The offsite consequences from shutdown risk are judged to be negligible on the following basis:~~

- ~~The significant shutdown events occur during Mode 6, which does not begin until approximately 96 hours after shutdown. The decay of fission products after 96 hours reduces the source term to less than 1% of the value at power operating conditions. Therefore, a postulated core damage event during shutdown would have a significantly lower source term and resultant offsite consequences than a containment bypass at full power.~~
- ~~The lower drywell hatches are only open for a limited period of time during Mode 6 to allow under vessel maintenance activities on the control rod drive mechanisms and neutron monitoring instrumentation. The details of exposure time are not developed in the design phase, but administrative controls will be implemented to limit the time that the hatches are open, as well as provide compensatory guidance if a line break occurs while the hatches are open. Therefore, the frequency of containment bypass events during shutdown can be significantly reduced.~~

<u>The source terms for containment bypass events may not fall below the early fatality threshold until approximately 8 days after shutdown; however, the frequency of shutdown containment bypass events is very low. As a result the offsite consequences, which are the product of the source term risk and the shutdown containment bypass frequency, are not significant.</u>

load, a sufficient quantity of water is available in the spent fuel pool to allow boiling for 72 hours and still provide acceptable fuel coverage in the pool. A dedicated external connection to the FAPCS line allows for manual hook-up of external water sources, if needed, at 7 days for either upper containment pool replenishment and for spent fuel pool makeup. These functions are manually actuated from the yard area and can be performed without any support systems.

The following components are within the scope of RTNSS, with the exception of those components described as safety-related in Tier 2 Subsection 9.1.3: the diesel-driven makeup pump system, FAPCS piping connecting to the diesel-driven makeup pump system, the external connection.

19A.3.1.3 *Control Room Habitability*

Safety-related portions of the Control Room Habitability Area Ventilation System maintain control room habitability. This function is operated on safety-related battery power for the first 72 hours following an event. For longer term operation, the system can be powered by a small, portable AC power generator that is kept on the plant site.

This generator is included within the scope of RTNSS.

19A.3.1.4 *Post-Accident Monitoring*

Operator actions are not required for successful operation of safety-related systems for the first 72 hours following an event. Beyond that, operator actions are necessary to support continued operation of decay heat removal and control room ventilation systems. These functions can be performed without any support systems or indications (other than local indications on the equipment to be operated).

However, the operators can use information on the condition of the plant to determine ways to augment the functions needed for beyond design basis response. This provides an additional flexibility (defense-in-depth) for the operators to respond in the post-72 hour time frame.

The Distributed Control and Instrumentation System (DCIS) that is powered by the safety-related power systems is used to perform this monitoring. In order to support monitoring beyond 72 hours, it is necessary to provide power for the Q-DCIS components. Two 6.9 kV Plant Investment Protection (PIP) nonsafety-related buses (PIP-A and PIP-B) provide power for the nonsafety-related PIP loads. PIP-A and PIP-B buses are each backed by a separate standby onsite AC power supply source. Cooling for the areas containing the DCIS components may also be required, depending on the outcome of the detailed building heatup analyses. These functions are provided by nonsafety-related SSCs that are candidates for RTNSS.

The standby diesel generators and the PIP buses provide power for Q-DCIS. Portions of the HVAC systems in the Reactor Building, Electrical Building, Fuel Building, Control Building, and some areas of the Turbine Building perform component and area cooling. In addition, support for these nonsafety-related functions is required from Reactor Component Cooling Water, Plant Service Water, and the Chilled Water System.

19A.3.2 Seismic Assessment

The seismic margins analysis described in section 19.2.3.~~5~~2.4 assesses the seismic ruggedness of safety-related plant systems and the non-safety systems required for decay heat removal after

72 hours. No accident sequence has a High Confidence for Low Probability of Failure (HCLPF) ratio less than 1.67 times the peak ground acceleration magnitude of the safe shutdown earthquake (SSE).

Therefore, there are no additional RTNSS candidates due to seismic events.

19A.4 CRITERION C: PRA MITIGATING SYSTEMS ASSESSMENT

Criterion C requires an assessment of safety functions that are relied upon at-power and during shutdown conditions to meet the NRC's safety goal guidelines. A comprehensive assessment to identify RTNSS candidates includes focused PRA sensitivity studies for internal events, evaluations of external events, an assessment of the effects of nonsafety-related systems on initiating event frequencies, and an assessment of uncertainties in these analyses and uncertainties that may be introduced by first of a kind passive components.

19A.4.1 Focused PRA Sensitivity Study

A focused PRA sensitivity study evaluates whether passive systems alone are adequate to meet the NRC safety goals of CDF less than 1.0 E-4 per year and LRF less than 1.0 E-6 per year. The focused PRA retains the same initiating event frequencies as the baseline PRA, and sets the status of nonsafety-related systems to failed, while safety-related systems remain unchanged in the model. The focused PRA model is evaluated using only the safety-related systems and RTNSS systems determined from criteria A or B. Additional nonsafety-related systems are included only if they are required to meet the CDF or LRF goals. The additional nonsafety-related systems required to meet the CDF and LRF goals are candidates for RTNSS.

The CDF and LRF goals will be met with the addition of portions of the Diverse Protection System (DPS) as RTNSS. This is needed to counter the effects of a dominant risk contribution due to common cause failures of actuation instrumentation and controls.

19A.4.2 Assessment of Non-Safety Systems on External Events

The effects of nonsafety-related systems relative to external events, at power and during shutdown, have a negligible effect on the CDF and LRF goals. The insights described in this subsection support this conclusion.

19A.4.2.1 *Fire*

The Fire PRA is a bounding analysis that incorporates several conservative assumptions. The fire analysis does not account for the amount of combustible material present, or for the distance between fire sources and targets. The analysis assumes that a fire ignition in any fire area grows into a fully developed fire. Therefore, fires are conservatively assumed to propagate unsuppressed in each fire area and damage all functions in the fire area.

The ESBWR probabilistic internal fire analysis highlights the following key insights regarding the fire mitigation capability of the ESBWR:

- The basic layout and safety design features of the ESBWR make it inherently capable of mitigating internal fires. Safety system redundancy and physical separation by fire