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MFN 07-011 Docket No. 52-010

January 26, 2007

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis – RAI Numbers 15.2-6 through 15.2-13, 15.3-8, 15.3-20, 15.3-21 and 15.3-23**

Enclosure 1 contains GE's response to the subject NRC RAIs transmitted via the Reference 1 letter.

If you have any questions or require additional information regarding the information provided here, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Project Manager, ESBWR Licensing

D068

Reference:

1. MFN 06-381, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 69 Related to the ESBWR Design Certification Application*, October 11, 2006

Enclosures:

1. MFN 07-011– Response to Portion of NRC Request for Additional Information Letter No. 69 – Safety Analysis – RAI Numbers 15.2-6 through 15.2-13, 15.3-8, 15.3-20, 15.3-21 and 15.3-23

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GE/San Jose (with enclosures)
	eDRF	0060-8006 – RAIs 15.2-6-9, 11-13
		0061-5483R1 – RAI 15.2-10
		0062-2740 – RAI 15.3-8
		0062-2465R1 – RAIs 15.3-20 and 21
		0062-8454 – RAI 15.3-23

Enclosure 1

MFN 07-011

Response to Portion of NRC Request for

Additional Information Letter No. 69

Related to ESBWR Design Certification Application

Safety Analysis

RAI Numbers 15.2-6 through 15.2-13, 15.3-8, 15.3-20, 15.3-21 and 15.3-23

NRC RAI 15.2-6:

DCD Tier 2, Rev. 1, Section 15.2.2 and elsewhere, states that the Steam bypass and pressure control (SB&PC) triplicated control system is not subject to a credible single failure. The statement seems to be based on circuitry and electronic operation of the system alone. However, mechanically, the system is mounted on a single device(s). Sections 15.2 and 15.3 emphasize the electronic control systems but ignore the possible contribution of the associated mechanical systems. For example, steam bypass or safety relief valves or control rods do not seem to contribute to the transient frequencies involving these components. A number of transients that could be initiated by SB&PC failure are in the anticipated operation occurrence (AOO) category. Have you included mechanical failure of the device(s)? A number of transients that could be initiated by SB&PC failure are in the anticipated operational occurrence (AOO) category. Please give the estimated failure frequencies of the triplicate control system. Are the electronic components themselves free of failures due to mechanical, heating, testing, vibration, and other causes?

GE Response:

It is correct that reliable control system design cannot preclude mechanical failures in the system being controlled. However, reliable control system design can significantly reduce the probability of failures that impact the entire system. In the case of the SB&PC system four Turbine Control Valves (TCVs) and twelve Turbine Bypass Valves (TBVs) are controlled. Single mechanical, or control system, failures can impact a single valve but only multiple failures causing failure of the SB&PC system can impact ALL valves. Therefore, we have analyzed a bounding subset of single valve events as AOOs (DCD Tier 2 Section 15.2) and a bounding subset of multiple failures that impact all valves as infrequent events (DCD Tier 2 Section 15.3). The evaluation of the bounding SB&PC system failure infrequent events is found in DCD Tier 2 Subsections 15.3.3 though 15.3.6. The associated evaluation of the failure frequencies of the triplicated control system are presented in DCD Tier 2 Subsections 15A.3.1 though 15A3.4. This frequency analysis does not consider multiple mechanical failures of all the valves to all open or all close at the same time because that probability is less than the probability of triplicated control system failures.

The response to RAI 15.2-8 illustrates that the failures considered in AOO analysis envelope single mechanical and control system failures expected in the TCVs and TBVs. Note that no frequency evaluation is provided for these failures because they are evaluated in the highest frequency category, AOO.

DCD Impact:

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

NRC RAI 15.2-7:

Per Standard Review Plan (SRP) Section 15.6.1, Revision 1, July 1981, TMI Action Item II.K.3.16, evaluation of the safety relief valve (SRV) performance, should be addressed in the DCD and the results should be included in the frequency evaluation and categorization of the inadvertent opening of an SRV event. Has this issue been addressed in the ESBWR DCD?

GE Response:

DCD Tier 2 Table 1A-1 discusses ESBWR resolution of TMI Action Item II.K.3.16. The probabilities of an inadvertent opening of an SRV and a stuck open SRV are evaluated in DCD Tier 2 Subsection 15A.3.8 and 15.A.3.10, respectively.

DCD Impact:

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

NRC RAI 15.2-8

In DCD Tier 2, Rev. 1, Section 15.2 (and elsewhere), the transients call for bypass and turbine control valves to open and close. Instrumentation failures are considered, but valve mechanical failure is not. Examples include failure to reseat or stuck closed valves, for which there exists a considerable database of experience. Address the issue of valve mechanical failures in the context of creating a non-analyzed condition or a new transient.

GE Response:

Although mechanical failure of the Turbine Control Valves (TCVs), Turbine Stop Valves (TSVs) and the Turbine Bypass Valves (TBVs) is not discussed specifically, the AOO events analyzed in DCD Tier 2 Section 15.2 envelop the mechanical failures of these valves. The discussion below addresses the mechanical failure modes of the TCVs, TSVs and the TBVs to show that they are analyzed as AOOs, or bounded by AOO analysis. One critical aspect of the design of the valves is that they are mechanically separate. The only connection between the valves is through the control systems. Detailed design of the valves and the turbine protection system is given in DCD Tier 2 Section 10.2 and Subsection 10.4.4.

Failure Modes Evaluation for TCVs during normal operation:

Normal operation is SB&PC system is controlling reactor pressure by moving the TCVs. At full power, during normal operation, the TCVs would be partially or full open. The TBVs would be fully closed and the TSVs would be fully open and not expected to move.

Fail Open: See DCD Tier 2 Subsection 15.2.5.1.

Fail Closed: See DCD Tier 2 Subsection 15.2.2.1. A TSV closure (0.1 s) is bounded by the fast closure time of a TCV (0.08 s) and therefore not analyzed separately.

Fail As-Is: If a TCV is stuck full or partially open, the SB&PC system will use the remaining valves to control pressure. If power increases and steam flow becomes larger that the three operating valves can pass, the SB&PC will open the bypass valves. There would be very small impact on reactor pressure and bounded by events evaluated in DCD Tier 2 Subsection 15.2.2. If power is reduced such that steam flow is less than that passed by the single stuck open valve (on the order of 20 % to 30 % power) and the operators did nothing about it, reactor pressure would start to reduce. At this point, the operators would likely take action. If the operators ignored the decreasing pressure the MSIVs would eventually close on low pressure (DCD Tier 2 Subsection 15.2.2) if the turbine protection system did not act to close the TCVs and TSVs first. Since this occurs at low power, the impact on thermal limits would be significantly decreased due to the thermal limit margin at low power.

Fail to Reseat: The TCVs are not expected to reseat under normal pressure control operation. If a valve fails to close on demand the result is the same as fail as-is.

Failure Modes Evaluation for TCVs and TSVs following turbine protection trips:

There are many turbine protection trips. These trips are designed to protect the plant investment not for reactor safety. These trips close the TCVs and the TSVs. These valves provide redundancy in shutting off flow to the turbine. The purpose for having the TSVs redundant to the TCVs is to ensure high reliability in shutting off steam flow to the turbine allowing for a single mechanical failure. See discussion in DCD Tier 2 Section 10.2.

Fail Open: The TCVs and TSVs are redundant, a failure of one valve to close would not change the event significantly. DCD Tier 2 Subsection 15.2.2 analyzes closure of the TCVs (load rejection) and the TSVs (turbine trip).

Fail Closed: Turbine protection trips demand closure of these valves.

Fail As-Is: Same as fail open (TSVs and TCVs are redundant).

Fail to Reseat: TSVs and TCVs are redundant. In any case, leakage past the valves is not a reactor safety concern.

Failure Modes Evaluation for TBVs:

The TBVs are normally closed. The TBVs serve to limit the pressure increase to the reactor following turbine protection trips and other anticipated operational occurrences. Only the case of a full closure of the TCVs/TSVs would require all the TBVs to open.

Fail Open: One valve could fail open. See DCD Tier 2 Subsection 15.2.5.1.

Fail Closed: The TBVs are normally closed this is the same as fail as-is.

Fail As-Is: Failure of a bypass valve to open still leaves 11 valves to perform the function. Failure of more valves to open is less likely, however, to clearly bound expected performance, DCD Tier 2 Subsections 15.2.2.3 and 15.2.2.5 assume half of the bypass vales fail on demand.

Fail to Reseat: Failure of the TBVs to reseat after demanded to open could impact condenser vacuum. If leakage was significant condenser vacuum could be lost. See DCD Tier 2 Subsection 15.2.2.8.

DCD Impact:

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

<u>NRC RAI 15.2-9</u>:

DCD Tier 2, Rev. 1, Tables 15.2-10 and 15.2-11 show turbine stop valve closing times of 0.16 and 0.10 seconds respectively. Conditions seem to be identical and the 0.16 is designated as "realistic closure timing." Provide explanation for using 0.10 seconds in DCD Tier 2, Table 15.2-11 and whether this is a realistic closure time. Explain the effect on the transient when a realistic closure time value is used.

<u>GE Response</u>:

The "realistic closure time" is expected to bound valve motion for a full bypass plant; however, the shorter closing time has been specified as a bounding value. The bounding value (0.1 for TSVs and 0.08 for TCVs) may result in slightly larger ΔCPR result. The load reject with bypass and the turbine trip with bypass will be run with the bounding closing time and the results will be presented in DCD Tier 2 Revision 3.

<u>DCD Impact</u>:

The load rejection run results with bypass and turbine trip with bypass will be provided in DCD Tier 2, Section 15.2 changes in Revision 3.

NRC RAI 15.2-10:

DCD Tier 2, Chapter 15 dismisses reactivity anomalies in the AOO category. The DCD does not include any justification that the control rod malfunctions are common (within or greater frequency than 10e-2). The argument has been made that the electronic portion of the control system has been improved. However, the mechanical part of control rod insertion/withdrawal is not mentioned. The Appendix 15A "Event Frequency Determination," sections 15A.3.11-13 regarding control rod errors during refueling, startup and operations, finds that inadvertent criticality to be at most 1.0d-7, 1.2e-6, and 1.5e-7 per RY, respectively. Such frequencies would qualify to be analyzed in the infrequent evenst or in the accidents section.

Section 15.3.7.2 states that "During refueling..interlocks provide assurance that inadvertent crirticality does not occur.." Likewise, section 15.3.9.2 concludes that "There is no basis for occurrence of the continuous control rod withdrawal error event in the power range." Yet, the probability estimates are in the same range as in the startup case for which some kind of analysis was provided. Justify the exclusion of reactivity anomalies from AOOs and include the mechanical part of the reliability including test data of the ESBWR control rod system. Was the difference in the estimated probability values of reactivity transients for refueling and power operation versus the startup the reason not to analyze refueling and power operation reactivity transients? If operational data was used in the estimation of the probability of reactivity transients, please describe the data used. Are the electronic components themselves free of failures due to mechanical, heating, testing, vibration, and other causes?

GE Response:

The ESBWR Fine Motion Control Rod Drives (FMCRDs) are distinguished from the locking piston control rod drives (LPCRD) in that the control rods are moved electrically during normal operation. The LPCRDs are used in most BWRs prior to the ABWR, which uses the FMCRD. The FMCRD and LPCRD are inserted into the core hydraulically during emergency shutdown. However, the FMCRD also has an electric motor to drive the control rod into the core even if the primary hydraulic system fails to do so.

Control rod insertion and withdrawal are controlled by the Rod Control and Information System (RC&IS). Some components of the RC&IS that involve mechanical type functions are relays (most of which are normally de-energized) and a few hard switches (pushbuttons in the control room). Mechanical failure of a single relay will not cause an inadvertent Rod Withdrawal Error (RWE). Additionally, failure of the contact of a single switch will not cause a RWE to occur. Therefore, the mechanically operated RC&IS equipment is single failure proof in regards to the RWE. Failure of electronic equipment in one channel of the redundant RC&IS equipment will not result in an inadvertent RWE, but could result in the inability to move the associated FMCRD by normal motor movement.

Several mechanical improvements to the CRD system are described below:

- The FMCRD is inserted hydraulically in response to a scram. The typical locking piston drives discharge the water from this type of hydraulic action in a scram discharge tank. This tank often causes maintenance and operational issues. The FMCRD discharges the volume of water to the reactor vessel, thus eliminating the common mode failure source of the scram discharge tank.

- The FMCRD pistons have no seals, thus do not require maintenance.
- A switch has been added to detect control rod separation. When the hollow piston is not properly seated on the ball nut or when the control rod separates from the hollow piston, a rod block is implemented.
- Latches have been added at axial intervals to prevent the control rod from dropping out of the core. There are also latches to hold the control rod full in after a scram until the ball-nut is run in to provide the normal support for the hollow piston and control rod.
- The control rod and hollow piston are coupled with a bayonet type coupling. This coupling is verified at refueling and during operation. The control rod can only be uncoupled from the FMCRD by relative rotation that is not possible during rotation as it is always constrained between four fuel assemblies.

Electrical components that would be involved for a RWE to occur would be the FMCRD motor and brake. These are non-safety related components but have been qualified for operability by FMCRD design life testing. The brake is environmentally and seismically qualified to provide the holding function when not energized. If RWE occurs, the brake is electrically energized by the RC&IS (Rod Brake Controller) equipment.

RWE during refueling is prevented by ensuring subcriticality due to rod withdrawal interlocks and the shutdown margin in any given core configuration. A discussion of event probability is given in Chapter 15A. Since criticality is prevented, RWE during refueling will not be analyzed. Shutdown margin calculations are presented in DCD Section 4.3. In addition to ensuring subcriticality, RC&IS logics normally prevent the withdrawal of more than one operable rod (i.e., all operable rods other than the one being withdrawn must be in full-in or rod block occurs). When in the SCRAM test mode of RC&IS, the RWM allows withdrawal of the two (or one rod for the central rod) rods associated with a single Hydraulic Control Unit (HCU). All other rods must be full-in for the movement of the two (or one) rods associated with one HCU to be withdrawn.

RWE at power is not analyzed due to the level of protection provided by the Rod Worth Minimizer (RWM) and Automated Thermal Limit Monitor (ATLM) subsystems of RC&IS that terminate any spurious rod movement prior to operating limit violation. There are two RC&IS channels. Any disagreement between the two initiates a rod block (unless one is bypassed). Any one channel can signal rod block.

Detection of an out-of-sequence movement when the reactor power is below the Low power setpoint by either channel of the RWM will cause as associated rod block to be enforced. If the serious failure of one channel of RWM equipment is detected with the reactor below Low Power Setpoint, with that channel not being bypassed, then a rod block is activated. The operator can bypass one channel of the RWM, but if the second channel is failed or bypassed then a rod block is activated. The operator can manually bypass one channel of the RWM; however, automatic control rod movement is prevented.

Above the Low Power Setpoint, the ATLM system monitors operating thermal limit protection function for either Minimum Critical Power Ratio (MCPR) or Maximum Linear Heat Generation

Rate (MLHGR). The protection algorithms block further control rod withdrawal when there is potential for either (MCPR or MLHGR) operating limit to be violated. If serious failure of one channel of ATLM equipment is detected with that channel not in bypass, a rod block is activated with RWM. The operator can bypass one channel of the ATLM, and if the second channel is failed or bypassed, a rod block is initiated.

These systems are discussed in DCD Tier 2 Subsection 7.7.2.

The data for electronic system failure and the beta factor used in frequency calculations for these events is from Reference 15.2-10-1 (Chapter 19, Table 19D.6-7, item Division 1 Transmission Network) as stated in DCD Tier 2 Subsection 15A.3.12.2.1. This reference documents the assessment of individual system failure.

It should be noted that the core response to this event is bounded by the control rod drop event described in DCD Tier 2 Subsection 15.4.6 and associated RAIs.

Reference:

15.2-10-1. GE Nuclear Energy, "23A6100, ABWR Standard Safety Analysis Report"

DCD Impact:

DCD Tier 2, Subsection 15.3.9.1 will be revised as noted on the markup below.

DCD Subsections 15.3.7-15.3.9 shall be moved to DCD Subsections 15.2.6-15.2.8.

15.3.9.1 Identification of Causes

The fine motion control rod drive moves on an electrical command from the RC&IS. Two dual channel subsystems of RC&IS, the Automated Thermal Limit Monitor (ATLM) and Rod Worth Minimizer (RWM), perform the associated rod block monitoring function. Any disagreement between the two initiates a rod block (unless one is bypassed). Any one channel can signal rod block.

Detection of an out-of-sequence movement when the reactor power is below the Low power setpoint by either channel of the RWM will cause as associated rod block to be enforced. If the serious failure of one channel of RWM equipment is detected with the reactor below Low Power Setpoint, with that channel not being bypassed, then a rod block is activated. The operator can bypass one channel of the RWM, but if the second channel is failed or bypassed then a rod block is activated. The operator can manually bypass one channel of the RWM; however, automatic control rod movement is prevented.

Above the Low Power Setpoint, the ATLM system monitors operating thermal limit protection function for either MCPR or MLHGR. The protection algorithms block further control rod withdrawal when there is potential for either (MCPR or MLHGR) operating limit to be violated. If serious failure of one channel of ATLM equipment is detected with that channel not being bypassed, then a rod block is activated as with RWM. The operator can bypass one channel of the ATLM and if the second channel is failed or bypassed then a rod block is initiated.

Detailed description of the ATLM subsystem is presented in Chapter 7.

The causes of a potential control rod withdrawal error are either a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. But in either case, the operating thermal limits rod block function blocks any further rod withdrawal when the operating thermal limit is reached. That is, the withdrawal of rods is stopped before the operating thermal limit is reached. Because there is no operating limit violation due to the preventive function of the ATLM, there is no rod withdrawal error transient event.

NRC RAI 15.2-11:

The calculated results of the transient resulting from inadvertent isolation condenser (IC) initiation are shown in DCD Tier 2, Rev. 1, Figure 15.2-11. In this figure (as well as Figure 15.2-1, depicting a similar transient) positive control reactivity is inserted at the same time as reactor power is increasing. In both instances, (but mainly in Figure 15.2-1) the minimum critical power ration (MCPR) gets close to or lower than 1.30. This action appears counter intuitive and appears to be the wrong thing to do. Explain why the system is designed to insert reactivity at that particular time.

GE Response:

The control rods do not move during the inadvertent isolation condenser initiation AOO. The control rod position is plotted in Figure 15.2-11f as "control fraction" (similar plots are provided for other AOOs and infrequent events). As can be seen on Figure 15.2-11f, the rods do not move during the transient. The change in control reactivity seen in Figure 15.2-11f is due to the manner in which control reactivity is calculated, not a sign of control rod movement.

The change in control reactivity is the summation of all the nodal reactivity differences between the controlled nodes and when these nodes are uncontrolled. This difference changes on a nodal basis as the flux conditions change from the effects of the transient. The control reactivity increase shown in the figure signifies that as a result of the transient the control rods in the core have less reactivity strength. Flux changes consider both the magnitude of the neutron flux and changes in the neutron energy spectrum.

DCD Impact:

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

NRC RAI 15.2-12:

The term "pump runout" implies excessive pump flow into lower pressure than the design pressure. DCD Tier 2, Rev. 1, Table 5.2-19 states "at system design pressure." Explain how the FW pump is able to increase its flow against design pressure for the Runout of One Feedwater Pump event.

GE Response:

The ESBWR reactor feedwater pumps are driven by an adjustable-speed, induction motor that is controlled by an adjustable speed drive. See DCD Tier 2 Subsections 7.7.3 and 10.4.7 for additional information. The term "runout" is used to refer to a failure in which a single feedwater pump is demanded to increase its speed. The following clarifications will be made to the discussion in DCD Tier 2 Section 15.2:

The following paragraph will replace the 1st paragraph of Subsection 15.2.4.2.1:

> *"The FW pumps (three normally operating) are driven by an adjustable-speed, induction motor that is controlled by an adjustable speed drive. This event is postulated on the basis of a single failure of a control device, specifically one that can directly cause an increase coolant inventory by increasing the speed of a single FW pump. The term "runout" is used in this section to describe this failure."*

The zero time event description in Table 15.2-19 will be revised to read:

> *"Initiate simulated increase in speed of one FW pump. The maximum pump flow is 75% at rated conditions."*

The value of 75% will bound actual performance. Actual pump capacity at maximum speed could be less.

DCD Impact:

Revision 3 to DCD Tier 2, Section 15.2 will reflect the above changes.

NRC RAI 15.2-13:

DCD Tier 2, Rev. 1, Section 15.2.4.2 includes extensive discussion of the improved electronics and conveys the impression that pump runout is a very low probability event. If this is the case, why is pump runout categorized in DCD Section 15.2 rather than Section 15.3?

GE Response:

The Feed Water Control System (FWCS) is highly reliable and failure of the FWCS to increase speed demand for ALL the FW pumps is considered an infrequent event. See DCD Tier 2 Subsection 15.3.2 for the event analysis and 15A.3.5 for event frequency determination. However, as discussed in the response to RAIs 15.2-6 and 15.2-8, the reliability of the control system is limited by the mechanical reality of the system being controlled. A single failure may cause an increase in speed of a single pump. DCD Tier 2 Subsection 15.2.4.2.1 paragraph six, discusses this possibility in more detail.

DCD Impact:

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

NRC RAI 15.3-8:

The FWCS is described in section 7.7.3.2 of the DCD, Tier 2, Rev. 1,and states that for power levels < 25 percent, it uses single element control with regards to operating modes. Does single element control provide the same degree of reliability as triple element operation? At normal power range operation the three-element control mode is utilized. Is there an "intermediate" (two element) control mode? What is the interface power level?

GE Response:

The ESBWR employs triple redundancy and either single or three-element control in the Feedwater Control System (FWCS). Triple redundancy is defined as a system configuration that uses three separate sensing elements and three control channels for a process. This arrangement increases system reliability because only 2 out of 3 signals are required to continue steady state operation.

The three-element control mode is defined as the control mode that employs three process parameters (reactor water level, steam flow rate and feedwater flow rate) to control the reactor water level. The steam and feedwater flow rates are used to determine the system flows mismatch, so that any change in either the steam or the feedwater flow rate is detected providing an anticipatory signal to the reactor feedwater control system (FWCS). During normal operation, the FWCS controls the reactor water level based on the reactor level signal, in addition to using the steam and the feedwater flow rates to provide quicker control response to the operating condition changes.

Due to lower accuracies of the steam and feedwater flow rate measuring elements and the slower response requirement at low reactor power levels (less than 25%), the three-element control is not used. Instead, a single-element control based on the reactor water level is used. However, the degree of reliability of the FWCS, in either the single-element mode or the three-element mode, is similar since both use the same level of triple redundancy in the measured signals and the same triple-redundant controller in the FWCS.

No two-element (intermediate) control mode exists in the FWCS.

DCD Impact:

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

NRC RAI 15.3-20:

RAI Summary: Justify not analyzing the malfunctions of the automated rod movement control system leading to inadvertent reactivity transients.

Text: DCD Tier 2, Rev. 1, Section 15.2 states that no inadvertent reactivity transients could be found. Section 15.3.8.1 of the DCD states that reactivity transients can be caused by "...malfunctions of the automated rod movement control system." Section 15.3.9 of the DCD states that: "There is no basis for occurrence of the continuous control rod withdrawal error event in the power range." Malfunctions are not controllable, thus, they could be part of the AOOs and/or any power level of operation and should be analyzed accordingly. Provide the basis for not analyzing the malfunctions leading to inadvertent reactivity transients and the inconsistency in the referenced sections.

GE Response:

Reactivity anomalies can be caused by operator error or by the automated rod movement system. DCD Tier 2, Table 15A-3 gives the probability of a rod withdrawal error due to either initiating error. The frequency of the events are given as:

Rod Withdrawal During Startup: 1 Event in 741,000 years

Rod Withdrawal During Power Operation: 1 Event in 40,900 years

Based on these frequencies, the events are categorized as infrequent events.

RAI 15.2-10 contains additional information on the mitigation of inadvertent rod withdrawal at power. Analysis of a rod withdrawal event during startup has been identified as a COL holder item.

DCD Impact:

DCD Tier 2, Subsection 15.2.3 will be revised as noted below:

> "Based on the probability for limiting reactivity and power distribution anomalies, there are no reactivity and power distribution anomaly AOOs identified for the ESBWR."

DCD Tier 2, Subsection 15.3.9.2 will be revised as noted below:

> "Due to an operator error or a malfunction of the automated rod withdrawal sequence control logic, a single control rod or a gang of control rods is withdrawn continuously. The ATLM operating thermal limit protection function of either the MCPR or MLHGR protection algorithms stops further control rod withdrawal when either operating limit is reached. As there will be no operating limit violations, there is no basis for occurrence of the continuous control rod withdrawal error event in the power range.
>
> No operator action is required to preclude this event, because the plant design as described above prevents its occurrence."

NRC RAI 15.3-21:

RAI Summary: Justify not analyzing the rod withdrawal error event.

Text: DCD Tier 2, Rev. 1, Sections 15.3.9.2/3 states that the plant design precludes the rod withdrawal error event from happening and therefore, there is no need to analyze this event. The description in the DCD does not allow the reviewer to conclude that the event is impossible and thus, an analysis is not needed, since the analysis is the means to decide whether the design is safe. Provide the justification to forgo the analysis and include discussion of the electronic and the mechanical aspects of the design in the justification. Justify that the Automated Thermal Limit Monitor system will never fail and the possibility of not removing the permissive for rod withdrawal. Provide the basis for the probability of the mechanical system failure used in your justification.

GE Response:

Reactivity anomalies can be caused by operator error or by the automated rod movement system. DCD Tier 2, Table 15A-3 gives the probability of a rod withdrawal error due to either initiating error. The frequency of the events are given as:

Rod Withdrawal During Refueling: 1 Event in 1,000 years

Rod Withdrawal During Startup: 1 Event in 741,000 years

Rod Withdrawal During Power Operation: 1 Event in 40,900 years

Based on these frequencies, the events are categorized as infrequent events.

Due to the interlocks utilized during refueling, criticality is not possible; thus this event is incredible. During power operation the ATLM and RWM function of the RC&IS enforce rod blocks should there be an inadvertent rod movement prior to thermal operating limit violation. For more information on the RC&IS functions for this event see RAI 15.2-10. Many safety features (including RWM, flux and period rod blocks and scrams) prevent and mitigate a rod withdrawal error during startup, however rod withdrawal during startup has been identified as a COL item to confirm safety functions mitigate the event as expected.

The RC&IS controls rod movement and mechanical failure in this system are considered for mechanical failures leading to inadvertent rod withdrawal. The probability of failure of the RC&IS components is given in Chapter 15A. For further description of the system and justification see RAI 15.2-10.

DCD Impact:

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

NRC RAI 15.3-23:

In existing power reactors, operating experience shows there have been several inadvertent SRV openings and particularly incidents of partial closure. For DCD Tier 2, Rev. 1, Section 15.3.13, how did GE figure that the probability for this occurrence is in the infrequent event category? Was the mechanical history of SRV performance accounted for? What are the mechanical/electronic (signal) improvements and associated databases to justify this categorization.

GE Response:

As stated in the introduction of DCD Tier 2, Chapter 15, Section 15.3 – Analysis of Infrequent Events, the determination of event frequency is provided in DCD Tier 2, Appendix 15A, and the categories of anticipated operational occurrence (AOO), in accordance with 10CFR 50 Appendix A, and infrequent event (IE). DCD Tier 2 Appendix 15A, Section 15A.1, specifically defines an IE as an event with an occurrence frequency of less than one per 100 (reactor) years. Table 15.0-1 provides a classification determination matrix for AOO, IE, Accident and Special Events. Table 15.0-2 provides a listing of the postulated events and their corresponding classification, and Table 15.0-5 provides the safety analysis acceptance criteria of the postulated IE for the ESBWR design.

The categorization in Subsection 15.3.13 of the inadvertent opening of a safety relief valve (IORV) as an IE is based on the probability analysis presented under Subsection 15A.3.8.2. The analysis provides details for determining how the IE definition applies to the IORV. The analysis result uses the sum of initiating event probabilities for the postulated initiators.

The analysis for IORV probability shares, in part, the probability of control system initiators from the analysis of the depressurization valve inadvertent actuation by the digital control system that is provided in Subsection 15A.3.9. Both analyzed events involve components controlled by the automatic depressurization system (ADS) logic. ADS is a part of the engineered safety feature (ESF) logics that is both generically and specifically described in DCD Tier 2 Chapter 7 under discussions of the Safety System Logic and Control (SSLC). SSLC architecture is addressed in Subsection 7.1.1.2.1. The Chapter 7 discussions of SSLC, ESF and ADS provide details of the improvements made for the ESBWR design that support the analysis for the low probability of an IORV event.

The request above goes further to also include the occurrence of stuck open relief valve (SORV) events, for which a probability analysis is provided in Subsection 15A.3.10, and the event summary description is under Subsection 15.3.15.

With respect to potential event challenges that might cause an SRV actuation, the ESBWR design reduces these potential challenges by use of the isolation condenser system (ICS) to respond to anticipated operational occurrences that cause a reactor vessel pressure increase (Reference: DCD Tier 2 Subsection 5.4.6.1.2). In addition, DCD Tier 2, Section 3.9 – Mechanical Systems and Components, provides information concerning the methods of analysis for seismic Category I components and supports. Under Subsection 3.9.1.4 – Consideration for Evaluation of Faulted Condition, are included the SRVs among components that are required to operate correctly under various conditions up to and including faulted plant conditions. The definition of plant conditions is found under Subsection 3.9.3.1.1. This subsection also includes a table that identifies a faulted plant condition as an extremely low probability event with an encounter probability, P, per reactor year in the range of 1x10e-04 > P > 1x10e-06.

The Section 3.9 design requirements have previously been translated into SRV design and procurement specification requirements, including a rigorous and extensive program for pre-acceptance qualification testing for the ABWR. The severity of the ABWR pre-acceptance test program is designed and specifically intended to incorporate lessons learned from the past years of commercial nuclear plant operation. The ABWR pre-acceptance test program was applied to several units of the ABWR prototype SRV, and includes sufficient valve actuation cycles to simulate significant service time compared to the planned 60-year plant life under various loads and environmental conditions. The pre-acceptance test program includes several stages of performance checks, leak checks and test valve inspections during the preconditioning test phase actuations prior to a final series of tests under simulated successive emergency condition actuations and a final faulted condition actuation. The acceptance criteria specifically require that after testing under the emergency conditions the test valve shall not exhibit any deformation or wear that would degrade its performance beyond the specification prescribed limits. The acceptance criteria for the simulated faulted condition tests requires that deformation or wear exhibited shall not preclude the valve from performing its design basis accident safety functions. The pre-acceptance test program provides a confidence level that the valve design is mechanically sound and reliable when compared against the mechanical performance history of relief valves used in the nuclear industry. The ABWR experience will be incorporated into the design and procurement specifications for the ESBWR SRVs.

DCD Impact:

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