

GE-Hitachi Nuclear Energy Americas LLC

James C. Kinsey
Project Manager, ESBWR Licensing

PO Box 780 M/C A-55
Wilmington, NC 28402-0780
USA

T 910 675 5057
F 910 362 5057
jim.kinsey@ge.com

MFN 07-336

Docket No. 52-010

June 22, 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 Nos. 76, 85, and 96, Related to ESBWR Design Certification
Application – RAI Numbers 7.1-42, 7.1-49, 7.7-4, 7.7-6, 7.9-5, 7.9-6,
7.9-11, 7.9-12, 7.9-13, 7.9-16**

Enclosures 1 and 2 contain GHNEA's response to the subject NRC RAIs transmitted via the referenced letters.

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

Sincerely,



James C. Kinsey
Project Manager, ESBWR Licensing

DC68

NRO

Reference:

1. MFN 06-388, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 76 Related to ESBWR Design Certification Application*, October 11, 2006
2. MFN 07-054, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 85 Related to ESBWR Design Certification Application*, January 19, 2007
3. MFN 07-231, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 96 Related to ESBWR Design Certification Application*, April 12, 2007

Enclosures:

1. MFN 07-336, Response to Portion of NRC Request for Additional Information Letter Nos. 76, 85, and 96 Related to ESBWR Design Certification Application - RAI Numbers 7.1-42, 7.1-49, 7.7-4, 7.7-6, 7.9-5, 7.9-6, 7.9-11, 7.9-12, 7.9-13, 7.9-16
2. MFN 07-336 – DCD Markups – RAI Numbers 7.7-4, 7.7-6, 7.9-5

cc: AE Cabbage USNRC (with enclosures)
RE Brown GHNEA/Wilmington (with enclosures)
GB Stramback GHNEA/San Jose (with enclosures)
eDRF 0000-0069-3314
0000-0068-4941
0000-0068-5308
0000-0068-9206
0000-0068-9206
0000-0066-2494
0000-0069-3208

MFN 07-336

Enclosure 1

**Response to NRC Request for Additional Information
Related to ESBWR Design Certification Application
DCD Sections 7.1, 7.7, and 7.9**

RAI Numbers

**7.1-42, 7.1-49, 7.7-4, 7.7-6, 7.9-5, 7.9-6, 7.9-11,
7.9-12, 7.9-13, and 7.9-16**

NRC RAI 7.1-42

Define what is meant by the "fault tolerant" features of both the Fault Tolerant Digital Controller and the SSLC architecture that is specified to be fault tolerant. Identify these features as software, hardware or both and describe the system responses to each type.

GE Response

The fault-tolerant architecture of the triple redundant controllers in the non safety-related distributed control and information system (N-DCIS) will tolerate any single hardware failure (and some dual failures) in the input-control-output loop without any interruption in the process being controlled. The architecture both provides hardware redundancy that tolerates a single failure without process interruption and provides the necessary self-diagnostics to detect, tolerate, and alarm for those single failures. The self-diagnostic functions are an integral part of the software operating system and include each main processor as well as each I/O module and communication module. The only operator response required is to address the diagnostic message by arranging for the repair/replacement of the faulted component. Summary discussions of the fault-tolerant N-DCIS architecture are found in DCD Revision 3, subsections 7.1.4, "N-DCIS General Description Summary," 7.1.4.2, "N-DCIS Nonsafety-Related Design Bases Summary," 7.1.5.1.2, "N-DCIS Nonsafety-Related Design Bases," 7.1.5.7, "N-DCIS Functions," 7.7.3, "Feedwater Control System," 7.7.5, "Steam Bypass and Pressure Control System," and Section 7.8, "Diverse Instrumentation and Control Systems."

The safety-related distributed control and information system (Q-DCIS) employs redundant power supplies and redundant communication paths within each division and can tolerate any single failure in these components without any interruption of the safety function. The only required operator response is as above. There are single failures possible within a safety-related division which would render it inoperable but the ESBWR Q-DCIS is designed such that two divisional failures can be tolerated without any interruption of the safety-related functions.

The fault-tolerant hardware and software features of the Q-DCIS are further described in:

- Triconex Topical Report 7286-545-1-A, "Qualification Summary Report," submitted via MFN 07-101 for SSLC/ESF, and
- NEDC-33288P, "Application of Nuclear Measurement Analysis and Control for a new BWR (NUMAC Platform Architecture)," submitted via MFN 07-160.

DCD/LTR Impact

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

NRC RAI 7.1-49

Provide a copy of Reference documents 8-1 through 8-17 as listed in NEDO-33288, Section 8. By letter MFN 07-160, dated March 21, 2007, GE submitted Topical Report NEDO-33288, Revision 0, "Application of Nuclear Measurement Analysis and Control (NUMAC) for ESBWR Reactor Trip System" for NRC review and approval. The staff needs all the related reference documents to complete their review. Please provide a copy of Reference documents 8-1 through 8-17 as listed in NEDO-33288, Section 8.

GE Response

The following references are found in section 8 of NEDO-33288, Revision 0:

Reference 8-1 "NEDC-32410P-A, 'Nuclear Measurement Analysis and Control Wide Range Neutron Monitoring System (NUMAC-PRNM), Retrofit Plus Option III Stability Trip Function,' Licensing Topic Report, GE Nuclear Energy, Class III (proprietary), October 1995" was reviewed and approved by the NRC via SER dated September 5, 1995.

Reference 8-2 "NEDO-31439-A, 'The Nuclear Measurement Analysis and Control Wide Range Neutron Monitoring System (NUMAC-WRNMS),' Licensing Topic Report, GE Nuclear Energy, Class I (non-proprietary), October 1990" was reviewed and approved by the NRC via SER dated October 3, 1990.

Reference 8-3 "NEDE-24362-1-P, Revision 1, 'General Electric Environmental Qualification Program,' General Electric Company, Class III (proprietary), January 1983." This reference information should be NEDE-24326-1-P, Revision 1 and will be corrected in a future revision of the LTR. As addressed in NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Design," dated July 1994, NEDE-24326-1-P was reviewed and approved by the NRC via SER dated October 23, 1983.

Reference 8-4 "EPRI TR-102323 (TR-1003697), "Guidelines for Electromagnetic Interference Testing of Power Plant Equipment", Revision 3, November 2004." This reference will be deleted in a later LTR revision because existing references to RG 1.180 Revision 1, Mil Std 461E and IEC 61000-4-X series are sufficient.

Reference 8-5 "NEDO-11209-04A, 'Nuclear Energy Business Group Boiling Water Reactor (BWR) Quality Assurance Program Description (Revision 4)', General Electric Company, December 31, 1982." The NRC has reviewed and approved "NEDO-11209-04A, 'Nuclear Energy Business Group Boiling Water Reactor (BWR) Quality Assurance Program Description (Revision 8)', General Electric Company, March 31, 1989" via SER dated December 20, 2000. Reference 8-5 in the LTR will be updated in a future revision to use revision 8 of NEDO-11209.

Reference 8-6 "26A6642, ESBWR Design Control Document, Tier 2" was submitted to the NRC via MFN-07-108, "General Electric Company - ESBWR Standard Plant Design - Revision 3 to Design Control Document - Tier 1 and Tier 2, Chapters 1 through 18," dated February 22, 2007.

Reference 8-7 "USNRC Regulatory Guide 1.97, Revision 4, Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants" is accessible from Agencywide Documents Access and Management System (ADAMS) using Accession Number ML061580448.

Reference 8-8 "NEDO-33294, 'ESBWR Safety Criteria for Instrumentation and Control Systems', Revision 0" is not required for design certification and has been withdrawn from submission to the NRC. MFN-07-265, "Updated Integrated Plan and Schedule – ESBWR Design Certification Application," dated June 1, 2007 lists the LTRs that are required for design certification. This reference will be deleted from the reference section in a future LTR revision.

Reference 8-9 "26A6641, ESBWR Design Control Document, Tier 1" was submitted to the NRC via MFN-07-108, "General Electric Company - ESBWR Standard Plant Design – Revision 3 to Design Control Document – Tier 1 and Tier 2, Chapters 1 through 18," dated February 22, 2007.

Reference 8-10 "MIL-STD-461D, Requirements for the Control of Electromagnetic Interference Emissions and Susceptibility" has been superseded by MIL-STD-461E, "Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment." A future revision of the LTR will delete reference to MIL-STD-461D and replace with MIL-STD-461E. Regulatory Guide 1.180, Revision 1 endorses MIL-STD-461E.

Reference 8-11 "MIL-STD-462D, Test Method Standard for Measurement of Electromagnetic Interference Characteristics" has been superseded by MIL-STD-461E, "Requirements for the Control of Electromagnetic Interference Characteristics of Subsystems and Equipment." A future revision of the LTR will delete reference to MIL-STD-462D and replace with MIL-STD-461E. Regulatory Guide 1.180, Revision 1 endorses MIL-STD-461E.

Reference 8-12 "IEC Standard 61000-4-X Series, Electromagnetic Compatibility for Industrial-Process Measurement and Control Equipment" is endorsed by Regulatory Guide 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems," Revision 1, dated October 2003.

Reference 8-13 "EPRI NP-5652, Guideline for the Utilization of Commercial Grade Items in Nuclear Safety Related Application (NCIG-07)" is endorsed by NRC Generic Letter 91-05, "Licensee Commercial-Grade Procurement and Dedication Programs," dated April 9, 1991.

Reference 8-14 "NEDO-33251, 'ESBWR I&C Defense-in-Depth and Diversity Report', Revision 0, July 2006" was submitted to the NRC via MFN-06-240, "ESBWR I&C Licensing Topical Report – NEDO-33251," dated July 24, 2006.

Reference 8-15 "GE Nuclear Energy, 'General Electric Instrument Setpoint Methodology', NEDC-31336P-A, Class III (proprietary), September 1996" was reviewed by the NRC via SER dated November 6, 1995.

Reference 8-16 "NEDO-33226, 'ESBWR Software Management Plan'" is scheduled to be submitted to the NRC by June 27, 2007 as indicated in Enclosure 1 of MFN-07-265, "Updated Integrated Plan and Schedule – ESBWR Design Certification Application," dated June 1, 2007.

Reference 8-17 "NEDO-33245, 'ESBWR Software Quality Assurance Plan'" is scheduled to be submitted to the NRC by June 27, 2007 as indicated in Enclosure 1 of MFN-07-265, "Updated Integrated Plan and Schedule – ESBWR Design Certification Application," dated June 1, 2007.

DCD/LTR Impact

There is no impact to the DCD as a result of this RAI.

NRC RAI 7.7-4

Explain functional necessity of RPV low temperature alarm.

In DCD, Tier 2, Revision 1, Section 7.7.1.5, it is stated that the reactor pressure vessel (RPV) temperature has a low temperature alarm in the control room. Please explain the functional requirement for a low temperature alarm.

GE Response

The alarm function is to provide a warning to the operators that a potential thermal stratification condition has developed within the vessel. The alarm is actually a high bottom head to reactor coolant temperature differential alarm and not a low temperature alarm.

DCD Impact

DCD Tier 2, Subsection 7.7.1.5 will be revised as noted in the attached markup (see Enclosure 2).

NRC RAI 7.7-6

(A) DCD, Tier 2, Revision 2, Section 15.3.7.2, "Sequence of Events and System Operation for the Control Rod Withdrawal Error During Refueling event", states that "when the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the REFUEL position, only one control rod can be withdrawn when the RC&IS [rod control and information system] SINGLE/GANG rod selection status is in the SINGLE rod selection mode. When the RC&IS SINGLE/GANG rod selection status is in the GANG rod selection mode, only one control rod pair with the same HCU [hydraulic control unit] may be withdrawn. Any attempt to withdraw an additional rod results in a rod block by the RC&IS interlock." DCD Tier 2, Revision 2, Section 7.7.2.2.7.4, "Rod Block Function," provides a list of the conditions which result in a rod block signal to the RC&IS. However, the rod block signals discussed in DCD Section 15.3.7.2 are missing from this list. Please revise DCD Tier 2, Section 7.7.2.2.7.4 to include these rod block interlocks and verify that all other rod block interlocks assumed in the analyses provided in DCD Tier 2, Chapter 15 are listed accordingly. Also, provide ITAAC in DCD Tier 1, Table 2.2.1-1, "ITAAC For Rod Control and Information System," for all rod block interlocks assumed in the analyses provided in DCD Tier 2 Chapter 15 analyses.

(B) DCD, Tier 2, Revision 2, Section 15.2.1.1, "Loss of Feedwater Heating," states that the Feedwater Control System (FWCS) Logic is provided in subsection 7.7.3, and includes logic to mitigate the effects of a loss of feedwater heating capability. However in DCD, Tier 2, Revision 2, Section 7.7.3, there is no such description of FWCS logic, and no logic diagram is provided. Therefore, in this respect, Chapter 7 & 15 are inconsistent. Please update DCD, Tier 2, Section 7.7.3 accordingly.

(C) DCD, Tier 2, Revision 2, Section 7.2.1.2.4.2, "Initiating Circuits," lists turbine stop valve (TCV) fast closure as "any one or more of the conditions listed below" that the "RPS logic initiates a reactor scram." DCD, Tier 2, Revision 2, Section 15.2.2, "Increase in Reactor Pressure" identifies several operational occurrences causing this and the resulting overall system responses. It states that if "the control system verifies that bypass capacity is adequate, the system will activate the SCRRI to reduce the power to 60 percent ". In other situations it explains, "an increase in system pressure and reactor shutdown (will happen) if the available turbine steam bypass capacity is insufficient." Therefore a scram is only imminent, after TCV fast closure, if there is insufficient steam bypass capability. Please address this apparent inconsistency between Chapter 7 and 15 and verify that a reactor scram following TCV fast closure would occur only if there is insufficient steam bypass capability.

(D) Inconsistencies have been previously identified by the staff between DCD Chapters 7 and 16. In light of these inconsistencies and the inconsistencies identified above between Chapters 7 and 15, please inform the staff of corrective actions that have been taken to ensure consistency within the DCD.

GE Response

(A) The single and dual control rod control interlocks and the rod block functions associated with improper movement of the single and dual control rods are described in DCD Tier 2, Revision 3, Subsection 7.7.2.2.7.6 "RC&IS Bypass Capabilities," as the Single/Dual Rod Sequence Restriction Override (S/DRSRO) bypass feature, and in Subsection 7.7.2.2.6, "RACS

Cabinets Subsystems," as the Rod Action Position Information (RAPI) trouble and the Rod Worth Minimizer (RWM) rod blocks.

Subsection 7.7.2.2.7.6 describes the S/DRSRO function:

"The RC&IS Single/Dual Rod Sequence Restriction Override (S/DRSRO) bypass feature allows the operator to perform special dual or single rod scram time surveillance testing at any power level of the reactor. In order to perform this test, it is often necessary to perform single or HCU pair rod movements that are not allowed normally by the sequence restrictions of the RC&IS.

When a control rod or pair of control rods associated with an individual HCU is placed in a S/DRSRO bypass condition, those control rod(s) are no longer used in determining compliance to the RC&IS sequence restrictions (for example, the ganged withdrawal sequence and RRPS).

The operator can only perform manual rod movements of control rods in the S/DRSRO bypass condition. The logic of the RC&IS allows this manual single/dual rod withdrawal for special scram time surveillance testing."

Subsection 7.7.2.2.6 describes the Rod Action and Position Information (RAPI) and the RWM functions. The functions germane to the issue are described as follows.

"The RAPI is the primary RC&IS equipment that performs the following functions:

- Enforces rod blocks based upon signals both internal and external to RC&IS. Internal RC&IS signals include those initiated from either of the two channels of rod blocks initiated by signals from the ATLM, RWM, RAPI SIU equipment, and those caused by any RAPI two-channel disagreement.
- Enforces adherence to a predetermined rod pull sequence that is stored in RRPS memory. The RRPS memory defines the order in which gangs of control rods are selected and moved when either semi-automatic or automatic rod movements are performed (i.e. the equivalent to the pull sheet used by plant operators when performing manual rod movements for conventional BWR plants)."

"The RWM issues a rod withdrawal block signal and a rod insertion block signal that is used in the RAPI rod block logic. This rod block signal ensures that:

- Absolute rod pattern restrictions called the ganged withdrawal sequence restrictions (GWSR), when reactor power is below the LPSP, are not violated (only applicable when the RPS reactor mode switch is in either STARTUP or RUN mode). The GWSR assure that control rod worths are maintained to within reasonable values by only allowing rod patterns that result in relatively low rod worths when control rods are withdrawn.
- Only the two control rods associated with the same HCU can be withdrawn for the 2-CRD scram time test when the RPS reactor mode switch is in the Refuel mode and the scram test mode has been activated. This function provides for performing individual HCU scram testing during planned refueling outages."

DCD Tier 2, Revision 3, Subsection 15.3.7.2 will be revised in DCD Tier 2, Revision 4, to clarify this linkage. See Enclosure 2 for the proposed revision of this section.

No additional rod blocks have been found within Chapter 15 that are not described in Chapter 7.

DCD Tier 1, Revision 3, Subsection 2.2.1, "Rod Control and Information System" describes these rod block features and includes appropriate ITAACs in Table 2.2.1-1. No changes to DCD Tier 1 will be made in response to NRC RAI 7.7-6(A).

(B) DCD Tier 2, Revision 3, Subsections 7.1.5.4.10, "Selected Control Rod Run In (SCRRI) / Select Rod Insert (SRI)" and 7.7.3.2.2, "Operation Modes" were revised to include descriptions of FWCS logic to mitigate the effects of a loss of feedwater heating Design Basis Event (DBE) by the Selected Control Rod Run-In (SCRRI) and Select Rod Insert (SRI) functions.

DCD Tier 2, Subsection 15.2.1.1, "Loss of Feedwater Heating," will be revised in DCD Tier 2, Revision 4, to clarify the FWCS linkage to N-DCIS, SCRRI and SRI functions. See Enclosure 2 for the proposed revision of this section.

(C) DCD Tier 2, Revision 3, Section 7.2, "Reactor Trip System" describes the scram initiating circuits and the automatic and manual bypass of selected scram functions in Subsections 7.2.1.2.4.2, "Initiating Circuits," and 7.2.1.14.2, "Automatic and Manual Bypass of Selected Scram Functions." The RPS operational bypasses are automatically controlled and inhibit actuation of those scram functions not required for a specific state of reactor operation in conformance with NRC Regulatory Guide (RG) 1.47 and IEEE Std. 603-1991, Criteria 6.6 and 7.4.

DCD Tier 2, Revision 3, Subsection 7.2.1.1.4.2.1, "Operational Bypasses," states, "the TSV closure and TCV fast closure reactor scram is automatically bypassed if a sufficient number of the bypass valves are opened (as indicated by their 10% position sensors) within a preset time delay after the initiation of the reactor trip signal caused by the TCV fast closure."

DCD Tier 2, Revision 3, Subsection 15.2.2, "Increase in Reactor Pressure," concludes that successful opening of a sufficient number of Turbine Bypass Valves and initiation of the SCRRI and SRI functions mitigates the Loss of Feedwater Heating, Generator Load Rejection, and Turbine Trip DBEs without requiring a reactor scram by the TCV fast closure signal; all other scram initiating parameters (i.e. RPV pressure, RPV level) will stay within their operating range.

(D) A number of inconsistencies were resolved during the development of DCD Tier 2, Revision 3, by comparing various DCD Tier 2 Chapters, such as Chapters 7, 15, and 16, to identify potential inconsistencies (e.g., terminology and logic functions). These potential inconsistencies were evaluated to determine the correct presentation and the appropriate changes were implemented. GE will continue to use this process in the development of subsequent DCD Tier 2 revisions.

DCD Impacts

DCD Tier 2, Subsections 15.3.7.2 and 15.2.1.1 will be revised in DCD Tier 2, Revision 4, as shown in Enclosure 2.

NRC RAI 7.9-5

Please confirm that all safety related electronic components including processors, video display units (VDUs), keyboard & mouse (if used), E to E (different divisions) gateways, E to NE gateways, I/O cards, cabinets, remote multiplexing units (RMUs), network interface modules, network communication modules, etc. are seismically qualified. What is the intended method of qualification for such electronic components and how does it conform to Regulatory Guide 1.100, Revision 2 - 06/1988. Provide a generic list of such electronic components.

GE Response

All safety-related electronic components including processors, video display units (VDUs), keyboard & mouse (if used), safety –related Distributed Control and Information System (Q-DCIS) to Q-DCIS (different divisions) gateways, Q-DCIS to nonsafety –related DCIS (N-DCIS) gateways (on the safety-related side), I/O cards, cabinets, remote multiplexing units (RMUs), network interface modules, network communication modules, etc. are qualified to seismic category 1 requirements (refer to DCD Revision 3, Subsection 7.1.6.6.1.5, Seismic Qualification discussion). A generic list of the major Q-DCIS components is provided in DCD Revision 3, Subsection 7.1.2.

As stated in DCD Revision 3, Subsection 7.1.6.6.1.5, qualification will be based on type testing in accordance with IEEE Std. 344, which conforms to the guidance of Regulatory Guide 1.100. Reference in this DCD Section to the 1975 version of IEEE Std. 344 will be deleted. The appropriate (1987) version of IEEE –344 is identified in DCD chapter 1 Table 1.9-22.

DCD/LTR Impact

DCD Subsection 7.1.6.6.1.5 Seismic Qualification will be revised to cite RG 1.100 (IEEE Std. 344), as shown in Enclosure 2.

NRC RAI 7.9-6

Regulatory Guide 1.12, Revision 2 - 03/1997, provides guidance for instrumentation to be provided to monitor the earthquake severity? How is this regulatory guide addressed?

GE Response

DCD Tier 2 Section 3.7.4, Revision 3, presents the ESBWR seismic monitoring system conformance with the guidance of Regulatory Guide 1.12, Revision 2, dated March 1997.

DCD/LTR Impact

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

NRC RAI 7.9-11

In DCD, Tier 2, Revision 1, Section 7.9.1.4, it is stated that D to A converters, if used in the remote multiplexing units (RMUs) will require periodic calibration check. It is expected that some of the controlled devices will be analog in nature (e.g., control valves)? Are such devices envisioned or only D to D interfaces considered even in modulating type control loops? If so, update DCD accordingly.

GE Response

In DCD Revision 3 Tier 2, Section 7.9.1.4 was re-drafted as Section 7.1.2.5, "Q-DCIS Testing and Inspection Requirements Summary." There are no digital to analog (D to A) converters in the ESBWR safety-related distributed control and information system (Q-DCIS). All such modulated outputs are found only in the nonsafety-related Distributed Control and Information System (N-DCIS). Digital to digital (D to D) interfaces to control analog devices are not included in the current design.

DCD/LTR Impact

DCD Revision 3 Subsection 7.1.2.5, last sentence, will be updated in a later revision by removing the parenthetical statement, "(also the digital-to-analog (D/A) converters (if used))".

NRC RAI 7.9-12

DCD, Tier 2, Revision 1, Section 7.9.1.5 states that "E-DCIS does not include sensor inputs up to the RMUs and RMU outputs to actuators." Please confirm that the output formats will be compatible with the actuated device and that this functionality is part of the essential distributed control and information system (E-DCIS), where applicable to the essential (E) devices.

GE Response

Although the RAI referenced statement was deleted when section 7.9 was combined with Section 7.1 in DCD Revision 3, it is restated in the rewritten section 7.1.3.2, "The field sensors and wiring belong to the process system to which they are attached and are not part of Q-DCIS".

The output formats will be compatible with the actuated device. This functionality is part of the safety-related Distributed Control and Information System (Q-DCIS).

DCD/LTR Impact

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

NRC RAI 7.9-13

DCD, Tier 2, Revision 1, Section 7.9.1.5 states that "When a network of the dual network system fails, operation continues automatically without operator intervention." Are there any time limitations to recover from failure of a single network?

GE Response

Both the safety-related Distributed Control and Information System (Q-DCIS) and the nonsafety-related Distributed Control and Information System (N-DCIS) employ dual redundant communication links. More specifically each of the four Q-DCIS divisions is an isolated network that uses dual power supplies and dual intra divisional communication links. Each Q-DCIS division is not completely single failure proof but the ESBWR design basis is that any two Q-DCIS divisions can be completely lost and all safety functions can still be performed. The N-DCIS is single failure proof including power supplies, controllers and communications.

In the event one of the dual Q-DCIS communication paths fails within a division, no safety functionality is lost and operation continues normally without any operator intervention (other than responding to the corresponding alarm). It is expected that the self-diagnostic capability of the Q-DCIS (and N-DCIS) hardware will allow quick repair/replacement of any failed component and that the network/division will be returned to service automatically with component replacement and/or power return. Other than the repair technicians, no operator action is required. There are no time limitations established for recovery from failure of a single network.

Likewise, with respect to the N-DCIS, if a network or network component fails, similarly no operator action is necessary, and no N-DCIS functionality will be lost. It is considered prudent to return the failed components to service in a reasonable time, but there is no safety issue associated with this failure. It should also be noted that the dual redundant N-DCIS network is not used for plant control other than operator input and monitoring; even if both networks failed, automatic ESBWR control systems would not be affected.

Where redundant communications between components/cabinets is employed, the failure of a single active transmitter/receiver or fiber will be alarmed, but will not interfere with the safety function; there is no recovery time limitation and the single failure can be repaired on line with or without bypassing the affected division. In the highly unlikely event that a double failure occurs in communications, an alarm is issued; the Nuclear Measurement Analysis and Control (NUMAC) Neutron Monitoring System (NMS) and Reactor Protection System (RPS) systems are "fail safe" and will interpret the double communications failure as a trip. The Safety System Logic and Control/Engineered Safety Features (SSLC/ESF) actuation instrumentation systems will fail "as is." This double failure would then constitute inoperability of the safety function and appropriate Technical Specification limitations would be imposed.

DCD/LTR Impact

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

NRC RAI 7.9-16

Please describe what type of provisions have been made for long term storage of historical data from safety as well as non-safety related systems and what provisions have been made for retrieving such data. Short-term data storage and retrieval and intervals up to 3 months are addressed for the non-essential distributed control and information system (NE-DCIS) historian in DCD, Tier 2, Revision 1, Section 7.9.2.1. Please confirm if any of the safety related data is stored in a safety related historian. If so, provide the capability of this part of the system.

GE Response

The system design is not yet complete; as such, specific details requested in this RAI are not available. However, as stated in DCD Revision 3, Tier 2, Section 7.1.5.5.9, the nonsafety-related Distributed Control and Information System (N-DCIS) includes a "historian" function that records measured and calculated point data for the plant, which includes safety-related and non safety-related data. It is expected that the N-DCIS storage capability will be enough to make this information available to the operator and plant engineering staff for analysis and trending for nominally a fuel cycle. As the N-DCIS local storage becomes full, the data can be archived on optical storage media (both locally and offsite) such that it will be available for later playback; the archiving will be possible on-line. The archived data can be played back as necessary on the plant N-DCIS or offline for easier analysis. It is not intended that safety-related data will be stored on a safety-related medium.

DCD/LTR Impact

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

MFN 07-336

Enclosure 2

DCD Markups

**RAI Numbers
7.7-4, 7.7-6, and 7.9-5**

RAI 7.7-4 Markups

ESBWR DCD Tier 2, Rev. 3, Chapter 7

7.7.1.5 Instrumentation Requirements

The following information is available to the reactor operator from the instrumentation discussed in this Subsection (IEEE Std. 603, Section 5.8):

- Reactor water level is indicated in the MCR on displays associated with the different water level ranges.
- The reactor pressure is indicated in the MCR and at four local racks in the containment.
- The discharge line temperatures of the SRVs are viewed on the VDUs in the MCR. Any temperature exceeding the trip setting is alarmed to indicate leakage of a SRV seat.
- RPV temperature is indicated and recorded in the MCR and low-high bottom head to reactor coolant differential temperature is alarmed in the MCR.

RAI 7.7-6 Markups

15.3.7.2 Sequence of Events and Systems Operation

Initial Control Rod Removal or Withdrawal

During refueling operation, system interlocks provide assurance that inadvertent criticality does not occur because a control rod (or a pair of control rods associated with the same HCU) has been removed or is withdrawn.

Fuel Insertion with Control Rod Withdrawn

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods be fully inserted when fuel is being loaded into the core. This requirement is backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the REFUEL position, the interlocks prevent the platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Likewise, if the refueling platform is over the core and fuel is on the hoist, control rod motion is blocked by the interlocks.

Second Control Rod Removal or Withdrawal

When the platform is not over the core (or fuel is not on the hoist) and the mode switch is in the REFUEL position, only one control rod can be withdrawn when the RC&IS SINGLE/GANG rod selection status is in the SINGLE rod selection mode. When the RC&IS SINGLE/GANG rod selection status is in the GANG rod selection mode, only one control rod pair with the same HCU may be withdrawn. The RC&IS Single/Dual Rod Sequence Restriction Override by-pass feature controls the movement of the control rods. Any attempt to withdraw an additional rod results in a rod block signal by the RC&IS interlock initiated by the RC&IS RAPI/RWM rod block logic. Because the core is designed to meet shutdown requirements with one control rod pair (with the same HCU) or one rod of maximum worth withdrawn, the core remains subcritical even with one rod or a rod pair associated with the same HCU withdrawn.

Control Rod Removal Without Fuel Removal

The design of the control rod, incorporating the bayonet coupling system does not physically permit the upward removal of the control rod without decoupling by rotation and the simultaneous or prior removal of the four adjacent fuel bundles.

Identification of Operator Actions

No operator actions are required to preclude this event, because the protection system design, as previously presented, prevents its occurrence.

- The automatic depressurization system (ADS) low water level setpoint analytical limit is 11.5 m above vessel 0.
- The maximum feedwater pump runout for a single pump is 75% of rated flow.
- For the transient representative of the loss of off site power with failure to transfer to internal power sources, it is assumed that initially a load rejection occurs, feedwater pumps trip and condensate pumps trip simultaneously.
- The stuck open safety relief valve transient has been analyzed with 4 ICs available and with a bounding capacity, to observe the maximum possible depressurization rate.

15.2.1 Decrease In Core Coolant Temperature

15.2.1.1 Loss Of Feedwater Heating

15.2.1.1.1 Identification of Causes

A feedwater (FW) heater can be lost in at least two ways:

- Steam extraction line to heater is closed; or
- FW is bypassed around heater.

The first case produces a gradual cooling of the FW. In the second case, the FW bypasses the heater and no heating of the FW occurs. In either case, the reactor vessel receives colder FW. The maximum number of FW heaters that can be tripped or bypassed by a single event represents the most severe event for analysis considerations.

The ESBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C (100°F) FW heating. The reference steam and power conversion system shown in Section 10.1 meets this requirement.

The loss of FW heating causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. However, the power increase is slow.

The Feedwater Control System (FWCS) ~~logic is provided in Subsection 7.7.3, and~~ includes logic to mitigate the effects of a loss of FW heating capability. The system is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected [i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint], the FWCS sends an alarm to the operator and sends a signal to the Nonsafety-Related Distributed Control and Information System (N-DCIS) ~~the Rod Control and Information System (RC&IS)~~ to initiate the Selected Control Rods Run-In (SCRRI) and Select Rods Insertion (SCRRI/SRI) functions to automatically reduce the reactor power and avoid a scram. This prevents the reactor from violating any thermal limits.

Control ~~blade rod~~ insertion is conservatively assumed to start only when the temperature difference setpoint is reached in the FW nozzle. The SCRRI/ and SRI functions ~~is are~~ able to suppress ~~totally~~ the neutron power increase and ensure that the MCPR reduction is small.

Events may exist where the SCRRI/ and SRI functions are ~~is~~ not activated; because the loss of feedwater temperature is less than 16.67°C (30°F). These events have a ΔCPR/ICPR similar to the event studied here, however none of them will become limiting.

15.2.1.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-4 lists the sequence of events for Figure 15.2-1

Because no scram occurs during this event, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems. The only assumed failure is for a single HCU, which actuates two control rods, avoiding the normal insertion of two rods.

15.2.1.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.

Results

Because the power increase during this event is controlled by the SCRRI/SRI insertion, the reduction of the MCPR is very small and is turned around when the SCRRI/SRI function takes effect. The results are summarized in Table 15.2-5.

No scram is assumed in this analysis. The increased core inlet subcooling aids thermal margins. Nuclear system pressure does not significantly change and consequently, the RCPB is not threatened.

This event is potentially limiting with respect to OLMCPR. The COL Applicant will provide reanalysis of this event for the specific initial and (COL Holder) reload core designs.

RAI 7.9-5 Markups

will be required to confirm that the maximum control room temperature plus mounting panel temperature rise, allowing for the heat load of the Q-DCIS equipment, does not exceed the temperature limit, and that control room humidity is maintained within limits.

Pressure: Q-DCIS components are designed to be qualified (by analysis) to perform to specification for any absolute pressure in the range specified. The design of the HVAC systems surrounding the Q-DCIS components ensures that the maximum control room pressure does not exceed the specified limit.

Radiation: Q-DCIS components are designed to be qualified (by analysis) to perform within specification limits over their service life under the specified radiation conditions. The design ensures that the maximum radiation levels at the equipment locations do not exceed the allowed limits.

Seismic Qualification: Q-DCIS components are designed to be qualified (by type testing and analysis) to demonstrate that the components will perform all specified functions correctly when operated within the specified seismic limits, and when mounted in accordance with the specified mounting methods. Q-DCIS components are to be qualified in accordance with the requirements of RG 1.100 (IEEE Std. 344 ~~1975~~). Qualification is based on type testing. The design ensures that the maximum seismic accelerations at the mounting locations of the equipment do not exceed the allowed limits.

EMI Qualification: Q-DCIS components in conformance with RG 1.180, when mounted in accordance with the specified mounting methods, are designed to be qualified by type testing and analysis to demonstrate that the components will perform all specified functions correctly when operated within the specified EMI limits. Q-DCIS equipment is designed to be not susceptible to electromagnetic disturbances from neighboring modules and does not cause electromagnetic disturbances to neighboring modules. The EMI qualification design follows the requirements specified in Mil Std. 461E and IEC 61000-4, depending on the specific requirement conditions. Q-DCIS equipment is qualified to perform within its specifications continuously while exposed to EMI environmental limits at the hardware mounting location. Reference 7.1-3 is used for the envelope limits. The EMI susceptibility and emissions testing is performed by type testing. In addition to the equipment design considerations, plant-specific actions are required to establish practices to control emission sources, maintain good grounding practices, and maintain equipment and cable separation.

7.1.6.6.1.6 System Integrity (IEEE Std. 603, Section 5.5)

Q-DCIS systems are required to accomplish their safety-related functions under the full range of applicable conditions enumerated in the design basis. Other areas addressed as requirements include adequate system real-time performance for digital computer-based systems to ensure completion of protective action, evaluation of hardware integrity and software integrity (software safety-related analysis, as part of BTP HICB-14 requirements), failure to a safe state upon loss of energy or adverse environmental conditions, and the requirements for manual reset.

Q-DCIS meets the integrity requirements described in IEEE Std. 603, Section 5.5. The RPS functions fail in the tripped state. The SSLC/ESF fails to a state such that the actuated component remains "as-is;" this prevents a control system induced LOCA. Hardware and