LICENSE AMENDMENT REQUEST TO CHANGE TECHNICAL SPECIFICATIONS IN SUPPORT OF PRNM AND ARTS / MELLLA IMPLEMENTATION
Enclosure 2 – Attachment 11

NEDO-33694, Revision 1

Columbia Generating Station Power Range Neutron Monitoring System Diversity and Defense-in-Depth (D3) Analysis

January 2012

(non-proprietary version)

GE Hitachi Nuclear Energy

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Columbia Generating Station Power Range Neutron Monitoring System Diversity and Defense-in-Depth (D3) Analysis

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Revision Summary

Revision	Change Summary
0	Initial Revision
1	Note 2 under Table 4-1 reworded to remove the discussion of the Rod Sequencing Control System, since it is not credited in Chapter 15 of FSAR for CGS. The paragraph is reworded to portray the description of FSAR Section 15.4.9.2.1. RSCS was removed from the acronym listing, since it is no longer used. Updated revision number in the reference of NEDC-33685P.

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ACRONYMS AND ABBREVIATIONS

Term	Definition
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
ATWS	Anticipated Transient Without Scram
BPWS	Banked Position Withdrawal Sequence
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CGS	Columbia Generating Station
CRDA	Control Rod Drop Accident
D3	Diversity and Defense-in-Depth
DBA	Design Basis Accident
DEH	Digital Electro-Hydraulic
ENW	Energy Northwest
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
FSAR	Final Safety Analysis Report
GEH	GE-Hitachi Nuclear Energy Americas LLC
GGNS	Grand Gulf Nuclear Station
HFE	Human Factors Engineering
HPCI	High Pressure Core Injection
HPCS	High Pressure Core Spray
IRM	Intermediate Range Monitor
IEEE	Institute of Electrical and Electronics Engineers
MSIV	Main Steam Isolation Valve
NBR	Nuclear Boiler Rating
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
OPRM	Oscillation Power Range Monitor
PLD	Programmable Logic Device

Term	Definition
PRNM	Power Range Neutron Monitoring
PRNMS	Power Range Neutron Monitoring System
PWR	Pressurized Water Reactor
RBM	Rod Block Monitor
RHR	Residual Heat Removal
RMCS	Reactor Manual Control System
RPS	Reactor Protection System
RTS	Reactor Trip System
RWE	Rod Withdrawal Error
RWM	Rod Worth Minimizer
SER	Safety Evaluation Report
SLO	Single Loop Operations
TCV	Turbine Control Valve
TLO	Two Loop Operations
V&V	Verification & Validation

EXECUTIVE SUMMARY

The Power Range Neutron Monitoring (PRNM) system upgrade was evaluated using the Acceptance Criteria identified in NRC Branch Technical Position (BTP) 7-19, Revision 6, *Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems*. This report provides an assessment of diversity and defense-in-depth, using the original Licensing Topical Report (LTR) (NEDC-32410P-A). Additionally, this report provides a detailed Diversity and Defense-in-Depth (D3) analysis based on a postulated worst-case common-cause failure in the PRNMS programmable entities, and directly addresses all criteria of BTP 7-19. The evaluations demonstrate that the plant has the diversity and defense-in-depth to cope with any potential common cause failure (CCF) in the programmable entities in the upgrade system.

1. INTRODUCTION

This report documents the Diversity and Defense-in-Depth (D3) analysis which has been performed for the Power Range Neutron Monitoring (PRNM) system upgrade at Columbia Generating Station (CGS). This report provides an assessment of D3 in Section 2, using the original Licensing Topical Report (LTR) (Reference 1). Additionally, within Sections 3 through 5, this report provides a worst-case D3 analysis, similar to the one previously presented to the U.S. Nuclear Regulatory Commission (NRC) for the Grand Gulf Nuclear Station (GGNS) PRNMS project, via responses to RAIs 8, 9 and 10 within Reference 2.

For the worst-case D3 analysis, the PRNMS is evaluated using the Acceptance Criteria identified in NRC Branch Technical Position (BTP) 7-19, Revision 6, *Guidance for Evaluation of Diversity and Defense-In-Depth in Digital Computer-Based Instrumentation and Control Systems* (Reference 3). The evaluation demonstrates that the plant has the ability to cope with any potential common cause failure (CCF) in the programmable entities in the upgrade system. Section 3 provides a description of the CCF to be used in the worst-case D3 analysis. Acceptance Criteria 1, 2, 6, 7, 8 and 9 relate to diversity and are discussed in Section 4; while Criteria 3, 4 and 5 relate to defense-in-depth and are discussed in Section 5.

2. ASSESSMENT USING LICENSING TOPICAL REPORT

The existing average power range monitor (APRM) subsystem provides a single-sensor input to the reactor protection system (RPS). Replacing the APRM subsystem with the PRNM system does not change or alter the diversity between RPS and the other plant systems that provide inputs to it. Other diverse sensors (e.g., reactor pressure) and manual RPS actuation provide adequate defense in depth to mitigate a CCF of the APRM subsystem. The PRNM system is the only Nuclear Measurement Analysis and Control (NUMAC) input into the RPS at CGS. The oscillation power range monitor (OPRM) is a single sensor input to RPS; the APRM and manual RPS actuation provide backup. GEH's approved design process and comprehensive Verification & Validation (V&V) program for the PRNM provide adequate reliability, including effects of possible software CCFs. This methodology, coupled with APRM and OPRM diverse functions and operator actions, provides an effective defense against potential CCFs in the PRNM system software.

2.1 Licensing Topical Report Information

An analysis of common cause software-related failures for the PRNM, which includes both APRM and OPRM functions, was previously performed by GEH (Reference 1) and approved by the NRC in their safety evaluation report (SER) (Reference 4). Relevant information from each document is presented below.

The conclusions of Section 6.5 of the PRNM LTR are applicable and CGS remains within its design bases. The design basis accidents and anticipated operational occurrences reported in the CGS Final Safety Analysis Report (FSAR) have been compared to those evaluated in the PRNM LTR. Events evaluated for the PRNM LTR encompass the events analyzed for CGS and the configuration of the PRNM is within the limits of the PRNM LTR.

CCF and Defense in Depth are covered for APRM and OPRM in Sections 6.4 and 6.5 of the PRNM LTR.

Regarding the APRM's function, Section 6.4.1, of Reference 1, references the analysis documented in GEH NEDC-30851P-A (Reference 5), which employs Electric Power Research Institute (EPRI) Report No. NP-2230, Part 3, Anticipated Transient Without Scram (ATWS): A Reappraisal: Frequency of Anticipated Transients. (Note: The NRC approved NEDC-30851P-A in a letter to the Boiling Water Reactor (BWR) Owners' Group dated January 24, 1988.)

Section 6.4.1 of the PRNM LTR states in part:

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Table F-1 is reproduced below as Table 2-1. Notes added to the table identify and resolve differences in the CGS design. The overall conclusions are that adequate diversity and defense in depth are provided, and that CGS's design is consistent with the PRNM LTR Section 6.4.1.

Table 2-1 Sensor Diversity for Initiating Events

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1 The CGS design does not include a scram on MSIV High Radiation. However, this design does not adversely affect the conclusions of NEDC-30851P-A as applied to CGS because other diverse RPS functions exist for the event that utilizes this scram as identified in the table.

2 CGS is also analyzed for these events without bypass capability. The scram sensors for the turbine and generator trip events are applicable regardless of bypass availability. Therefore, the diverse sensors identified for the "with bypass" events also apply to the "without bypass" events.

Regarding the OPRM function, Section 6.4.2 of the PRNM LTR states:

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Section 6.5 of the PRNM LTR documented the following conclusions:

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2.2 NRC Safety Evaluation Report

Section 6.6 of the PRNM LTR states the licensee must confirm applicability of these conclusions by:

(1) Confirming the events, defined in EPRI Report No. NP-2230 or Appendices F and G of NEDC-30851P-A, encompass the events that are analyzed for the plant;

(2) Confirming the configuration implemented by the plant is within the limits described in the PRNM LTR; and

(3) Preparing a plant-specific 10 CFR 50.59 evaluation of the modification per applicable plant procedures.

Energy Northwest (ENW) confirms Items (1) and (2) as follows:

(1) Table 2-2 demonstrates that the events defined in Appendices F and G of NEDC-30851P-A encompasses the events that are analyzed for CGS. Table 2-2 lists the events identified in Appendices F and G of NEDC-30851P-A and identifies the applicable section in Chapter 15, *Accident Analyses*, of the CGS FSAR in which the event is discussed.

(2) The CGS-specific PRNM System configuration is described in NEDC-33685P (Reference 6) System Description to the Block Diagram Level, which shows it is within the limits described in the PRNM LTR.

(3) The requirements of 10CFR50.59 have been applied to the PRNM modification in accordance with applicable plant procedures.

The NRC evaluated the PRNM System for common-cause software-related failures documented in the PRNM LTR and agreed with GEH's conclusions, as documented in its safety evaluation report (SER) approving the PRNM LTR. Specifically, Section 3.4.6 of the SER states:

> "GE performed equipment failure analyses to evaluate the effects of module level failures on critical system functions, and to assess qualitatively the defense-in-depth of the PRNMS. Common cause software related failures, which can result in PRNMS malfunctions were evaluated in the GE analyses. Defense-in-depth design features in the existing RPS, including the diverse anticipated transient without scram mitigation system and manual reactor trip capability, provide an acceptable means to address common mode failures in the APRM and OPRM software functions. Additionally, as mentioned above {Section 3.2 of the SER}, the APRM and OPRM software development process involves a comprehensive quality assurance methodology to detect and correct software errors. This

methodology, coupled with APRM diverse functions and operator actions, provides an effective defense against CCFs in the software. The staff finds the above features to address malfunctions to be acceptable."

Table 2-2 Cross-Reference of NEDC-30851P-A Events to CGS FSAR

Identified Event	***FSAR Section***
Appendix F – Transient/Accidents Analyses	
MSIV Closure	15.2.4
Turbine Trip (with bypass) (See Note 1)	15.2.3
Generator Trip (with bypass) (See Note 1)	15.2.2
Pressure Regulator Failure (Primary Pressure Decrease) (MSIV Closure)	15.1.3
Pressure Regulator Failure (Primary Pressure Decrease) (Level 8 Trip)	15.1.3
Pressure Regulator Failure (Primary Pressure Increase)	15.2.1
Feedwater Control Failure (High Reactor Water Level)	15.1.2
Feedwater Flow Control Failure (Low Reactor Water Level)	15.2.7
Loss of Condenser Vacuum	15.2.5
Loss of AC Power (Loss of Grid Connections)	15.2.6
Loss of AC Power (Loss of Transformer)	15.2.6
Appendix G – Other Events	
Loss Of One Feedwater Heater	15.1.1
Start of Idle Recirculation Pump between 60% and 65% CTP	15.4.4
Rod Withdrawal Error from 0% to 100% CTP	15.4.1, 15.4.2
Recirculation Pump Trip (One or Two Pumps)	15.3.1

Identified Event	*FSAR Section*
Loss of Instrument Air	7.3.2 7.4.2 Note 2
Recirculation Flow Control Failure (Increase Flow)	15.4.5
Recirculation Flow Control Failure (Decreasing Flow)	15.3.2
Inadvertent Opening of One Safety/Relief Valve	15.1.4
Inadvertent Residual Heat Removal (RHR) Shutdown Cooling Operations	15.1.6
Inadvertent Closure of One MSIV	15.2.4
Partial MSIV Closure	15.2.4
Recirculation Pump Seizure	15.3.3
Rod Withdrawal at Power	15.4.2, 15.4.9
High Flux due to Rod Withdrawal at Startup	15.4.1, 15.4.9
Inadvertent Insertion of Control Rods	Note 2
Detected Fault in RPS	Note 2
Inadvertent startup of High Pressure Core Injection (HPCI)/High Pressure Core Spray (HPCS)	15.5.1
Scram due to Plant Occurrences (Manual Scram)	Note 2
Spurious Trip via Instrumentation, RPS Fault	Note 2
Manual Scram – No Out-of-Tolerance Condition	Note 2

Note 1: CGS is also analyzed for this event without bypass capability, which is discussed in the referenced FSAR section.

Note 2: This event does not encroach upon any safety limit and as such is not specifically identified in the FSAR. The design and licensing basis for CGS continues to be met for this event as it is bounded by more limiting anticipated operational occurrences (AOOs) described in the FSAR.

2.3 Conclusion

The PRNM System replaces a single-sensor input to the RPS, but does not change or alter the plant-level diversity between RPS and other plant systems. Other sensor inputs within RPS (e.g., reactor dome pressure) are diverse from the PRNM System because these (other) sensor inputs do not utilize the NUMAC platform. Therefore, they are not subject to the same common-cause failures.

The APRM is a single sensor input to RPS; other diverse sensors (e.g., reactor pressure) and manual RPS actuation provide adequate defense in depth to mitigate a CCF of the APRM. The OPRM is a single sensor input to RPS; the APRM and manual RPS actuation provide backup. GEH's approved design process and comprehensive V&V program for the PRNM, provide adequate reliability including effects of possible software CCFs. This methodology, coupled with APRM and OPRM diverse functions and operator actions, provides an effective defense against potential CCFs in the PRNMS software.

3. DESCRIPTION OF POSTULATED WORST-CASE CCF FOR DETAILED (BTP 7-19) D3 ANALYSIS

A worst-case CCF in the PRNMS is postulated in order to perform the D3 assessment. Rather than postulating individual CCFs in each of the programmable entities ([[

]]), a single CCF that completely impairs the PRNM system is assumed. The postulated CCF in the PRNM system is assumed to remain latent and non-detectable until the system is stressed by an event or accident, at which time all PRNM system outputs from all four channels are absent or incorrect. In other words, the system is assumed to provide no advanced notice of trouble, fail to provide the correct responses such as rod blocks and trips during a transient, and also to provide misleading indications of plant parameters during the transient.

Different CCFs could have been postulated to occur in the 2-Out-Of-4 Logic Modules, or in the APRM instruments. Each of these scenarios is less severe than the worst-case CCF assumed. If a CCF occurs in the 2-Out-Of-4 Logic Modules, [[

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If a CCF in only the APRM instrument occurs, [[

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Additionally, CCFs in either the 2-Out-Of-4 Logic Modules or the APRM instruments [[

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Based on this reasoning, GEH maintains that the postulated CCF is a very remote scenario. Additionally, the NUMAC PRNMS has over 200 plant years of operation to its credit, and GEH is not aware of a single instance of a system that failed in this manner. Nevertheless, the single worst-case CCF in the PRNMS as described above is assumed in PRNMS.

4. BTP 7-19 DIVERSITY ANALYSIS

4.1 Analysis Approach

NUREG/CR-6303, *Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems* (Reference 7) Section 3.2, identifies and describes six types of diversity: design diversity, equipment diversity, functional diversity, human diversity, signal diversity, and software diversity. In cases where a diverse system is identified to respond in the absence of a response from PRNM system, the justification for evaluating the system as diverse is provided.

4.2 Evaluation

BTP 7-19 Acceptance Criteria (1) and (2)

(1) For each anticipated operational occurrence in the design basis occurring in conjunction with each single postulated CCF, the plant response calculated using realistic assumptions (e.g., plant operating at normal power levels, temperatures, pressures, flows, normal alignments of equipment, etc.) analyses should not result in radiation release exceeding 10 percent of the applicable siting dose guideline values or violation of the integrity of the primary coolant pressure boundary. The applicant/licensee should (1) demonstrate that sufficient diversity exists to achieve these goals, (2) identify the vulnerabilities discovered and the corrective actions taken, or (3) identify the vulnerabilities discovered and provide a documented basis that justifies taking no action.

(2) For each postulated accident in the design basis occurring in conjunction with each single postulated CCF, the plant response calculated using realistic assumptions analyses should not result in radiation release exceeding the applicable siting dose guideline values, violation of the integrity of the primary coolant pressure boundary, or violation of the integrity of the containment (i.e., exceeding coolant system or containment design limits). The applicant/licensee should (1) demonstrate that sufficient diversity exists to achieve these goals, (2) identify the vulnerabilities discovered and the corrective actions taken, or (3) identify the vulnerabilities discovered and provide a documented basis that justifies taking no action.

These first two criteria require an evaluation of each AOO and design basis accident (DBA), assuming the CCF in PRNM system occurs. The purpose of the evaluation is to ensure that sufficient diversity exists to allow the plant to cope with the events if they occur in conjunction with the postulated CCF. Table 4-1 lists each scenario from Chapter 15 of the CGS FSAR (Reference 8) and the credited trip response, if any. The right-most column contains the discussion or evaluation of the effect of the postulated CCF in the PRNMS. The conclusion is that there are no events that lead to any threat to the specified limits.

Table 4-1 Analysis of CGS AOOs and DBAs for Diversity

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.1.1	Loss of Feedwater Heating	None	No effect from postulated CCF because the analysis does not take credit for any PRNMS response. The current analysis does not take credit for any reactor trip.
15.1.2	Feedwater Controller Failure – Maximum Demand	During normal operations – Reactor High Water Level (L8) for Turbine Trip; Turbine Control Valve (TCV) Fast Closure for Reactor Trip; During single loop operations (SLO) – Reactor High Water Level (L8) for Turbine Trip; Turbine Stop Valve Closure for Reactor Trip	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.1.3	Pressure Regulator Failure - Open	Reactor High Water Level (L8) for Turbine Trip; Turbine Stop Valve Closure for Reactor Trip	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.1.4	Inadvertent Safety/Relief Valve Opening	None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.1.5	Spectrum of Steam System Piping Failures Inside and Outside of Containment in a Pressurized Water Reactor (PWR)	None	The event is not applicable to BWR plants.
15.1.6	Inadvertent RHR Shutdown Cooling Operation	None	No effect from a PRNM system CCF. The PRNM system is not credited in the analysis, but rather mentioned as the back-up to the primary protection, which is operator action. The event is applicable only during Startup or cool down operation, when Intermediate Range Monitor (IRM) also is in operation. The IRM is an analog system. Therefore, it is diverse from the digital PRNMS and not vulnerable to the postulated CCF.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.2.1	Pressure Regulator Failure	None	This event is classified as a moderate frequency event. FSAR 15.2.1.1 identifies that the moderate frequency event considered for this transient analysis is that of a single failure which occurs on the controlling pressure transmitter which erroneously causes the Digital Electro Hydraulic (DEH) control system to close the turbine control (governor) valves and thereby increases reactor pressure. FSAR section 15.2.1.2.1 specifies in the sequence of events that failure of a DEH control system component that causes the turbine control (governor) valves or turbine bypass valves to move towards the closed position will momentarily result in an initial pressure increase because the reactor is still generating the initial steam flow. The DEH control system is self-diagnostic and will detect the faulty component and disable it. The redundant control system will continue to perform its functions, and will restore steady state operations. The plant does not trip, and therefore there is no effect from a postulated CCF in PRNMS on this event.
15.2.2	Generator Load Rejection (with Bypass System operational)	Power > 30% NBR - Turbine Control Valve (TCV) Fast Closure, Power < 30% NBR - None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.2.2	Generator Load Rejection (with Bypass System failure)	Power > 30% NBR - Turbine Control Valve (TCV) Fast Closure, Power < 30% NBR - None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.2.3	Turbine Trip (with Bypass System operational)	Power > 30% NBR - Turbine Stop Valve Closure Power < 30% NBR - None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.2.3	Turbine Trip (with Bypass System failure)	Power > 30% NBR - Turbine Stop Valve Closure Power < 30% NBR - Hi Vessel Pressure	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.2.4	MSIV Closures – all valves	MSIV Closure	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.2.4	MSIV Closure – one valve	Neutron Flux, Vessel Dome Pressure	An automated response from a diverse safety-related system exists if PRNM system fails to respond. The Vessel Dome Pressure scram signal is issued by an analog system. Therefore, it is diverse from the digital PRNM system and not vulnerable to the postulated CCF.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.2.5	Loss of Condenser Vacuum	Turbine Stop Valve Closure	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.2.6	Loss of AC Power	Loss of Power to scram and MSIV Solenoids, Turbine Control Valve (TCV) Fast Closure	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.2.7	Loss of Feedwater Flow	Low Water Level (L3)	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.2.8	Feedwater Line Break	NA	NA - refer to 15.6.6.
15.2.9	Failure of RHR Shutdown Cooling	None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.3.1	Recirculation Pump Trip	1 pump – None 2 pumps – Reactor High Water Level (L8) for Turbine Trip; Turbine Stop Valve Closure for Reactor Trip	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.3.2	Recirculation Flow Control Failure - Decreasing Flow	1 pump – None 2 pumps – Reactor High Water Level (L8) for Turbine Trip; Turbine Stop Valve Closure for Reactor Trip	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.3.3	Recirculation Pump Seizure	During two loop operations (TLO) - Reactor High Water Level (L8) for Turbine Trip; Turbine Stop Valve Closure for Reactor Trip During SLO - None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.3.4	Recirculation Pump Shaft Break	Reactor High Water Level (L8) for Turbine Trip; Turbine Stop Valve Closure for Reactor Trip	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.4.1	Rod Withdrawal Error - Low Power	None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.4.2	Rod Withdrawal Error at Power	None	A PRNM system CCF would result in an unblocked Rod Withdrawal Error (RWE) event. The radiological consequence of this event is bounded by the radiological consequence of the Control Rod Drop Accident (CRDA). The CRDA is a rapid, uncontrolled control rod withdrawal that occurs at low power and has very little void feedback. It assumes fuel melt occurs for determination of the source term. The unblocked RWE is a slower event with significant void feedback and no fuel melting. The radiological consequence of the unblocked RWE is bounded by the CRDA and is within Acceptance Criteria (1) of BTP 7-19.
15.4.3	Control Rod Maloperation (System Malfunction or Operator Error)	NA	NA – refer to 15.4.1 and 15.4.2.
15.4.4	Startup of Idle Recirculation Pump	None	No threat to applicable limits is posed by PRNM system CCF. No protection systems' response is anticipated because the intent is to start the pump without a scram. Normal procedures prohibit starting the pump at a power level that would lead to automatic actions. Even if it were supposed that an operator error occurs and the pump is started when power is high enough for a neutron flux scram to occur during the flux spike, the analysis shows that the thermal power would rise more slowly and steady out at an acceptable higher level.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.4.5	Recirculation Flow Control Failure with Increasing Flow	Slow opening – None Fast opening – Neutron Flux	No threat to applicable limits is posed by PRNM system CCF. The slow opening of one recirculation flow control valve establishes the thermal limits basis for this event because the analysis process, which does not take credit for a scram during slow flow run-up, is designed to maximize the heat flux change. During a fast run-up event, a neutron flux scram may occur during the flux spike. If the scram is postulated to not occur, the heat flux after the event stabilizes would be similar to the slow run-up analysis, which imposes larger heat flux changes with no scram. This result is within Acceptance Criteria (1) of BTP 7-19.
15.4.6	Chemical and Volume Control System Malfunctions	NA	The event is not applicable to BWR plants.
15.4.7	Misplaced Bundle Accident	None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.4.8	Spectrum of Rod Ejection Assemblies	NA	The event is not applicable to BWR plants.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.4.9	Control Rod Drop Accident (CRDA)	Neutron Flux	An automated response from a diverse safety-related system exists if PRNM system fails to respond. The CRDA analysis ignored IRM for conservatism, but in reality, the IRM would terminate the event. The IRM is an analog system. Therefore it is diverse from the digital PRNM system, and not vulnerable to the postulated CCF. (Note 2) This result is within Acceptance Criteria (2) of BTP 7-19.
15.5.1	Inadvertent HPCS Startup	None, but Reactor Water Level (L3) is considered a backup for decreasing level. Reactor Water Level (L8) for Turbine Trip is considered a backup for increasing level.	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.5.2	Chemical Volume Control System Malfunction (or Operator Error)	NA	The event is not applicable to BWR plants.
15.5.3	BWR Transients Which Increase Reactor Coolant Inventory	NA	Refer to 15.1 and 15.2.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.6.1	Inadvertent Safety/Relief Valve Opening	NA	Refer to 15.1.4.
15.6.2	Instrument Line Pipe Break	Manual Scram after 20 Min	No effect from postulated CCF because the analysis does not take credit for any PRNMS response.
15.6.3	Steam Generator Tube Failure	NA	The event is not applicable to BWR plants.
15.6.4	Steam System Piping Break Outside Containment	MSIV Closure	No effect from postulated CCF because the analysis does not take credit for any PRNMS response.
15.6.5	Loss-of-Coolant Accidents (Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary - Inside Containment)	Low Water Level (L3) or High Drywell Pressure (Note 1)	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.6.6	Feedwater Line Break-Outside Containment	Low Water Level (L3)	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.7.1	Radioactive Gas System Leak or Failure	NA	This event is not applicable to CGS.
15.7.2	Liquid Radioactive System Failure	NA	This event is not applicable to CGS.

FSAR Section	Title	Credited Trip Signals	Evaluation / Discussion
15.7.3	Postulated Radioactive Releases Due to Liquid Radwaste Tank Failure	None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.7.4	Fuel Handling Accident	None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.7.5	Spent Fuel Cask Drop Accidents	None	No effect from postulated CCF because the analysis does not take credit for any PRNM system response.
15.8.0 – 15.8.11	Capabilities of Present Design to Accommodate ATWS	None	Per Section 15.8.0, "Anticipated Transients Without Scram (ATWS) events described in this section are not design basis events for CGS." This excludes ATWS from consideration for BTP 7-19 Acceptance Criteria (1) and (2).

Notes:

1. Obtained credited trips from Section 7.2.1.1.3 of the CGS FSAR (Reference 8).

2. The CRDA as analyzed and reported in Chapter 15 of the CGS FSAR takes a conservative approach that the event is terminated with an APRM High Neutron Flux Scram. The purpose of this approach is to provide a bounding analysis for the CRDA with the assumption that the reactor is past STARTUP mode (in RUN mode), with no credit for moderator feedback and minimal credit for the Doppler feedback and scram with the APRM High Neutron Flux Scram.

   The consequences of a CRDA are most severe in terms of fuel enthalpy increases during lower power STARTUP mode conditions. The APRM High Neutron Flux Scram would not be operational but rather the APRM Setdown Scram. The IRM scram would be the most likely scram signal. [[

   ]] Therefore, for a realistic CRDA scenario, a postulated CCF in the software of the PRNM preventing the scram from the APRM High Neutron Flux (or the APRM Setdown) would not affect the scram from IRM, the primary initiation signal for reactor trip.

Per Section 15.4.9.2.1 of the FSAR (Reference 8), the event is further mitigated by an initial control rod configuration that complies with the Banked Position Withdrawal Sequence (BPWS). The withdrawal (or insertion) sequence is implemented by the operator and enforced by the Rod Worth Minimizer (RWM). An operator error in control rod movement will be detected and stopped by the RWM. If the RWM system is not operable, rod movement can only continue with a backup for the operator verifying compliance with the BPWS sequence. Failure of the RWM concurrent with an operator error of moving an out-of-sequence rod, contrary to procedures would be required to result in a potentially more limiting event.

In realistic 3D evaluations for the limiting CRDA scenarios (Reference 11), the presence or absence of a scram is irrelevant because the effect of the scram occurs too late to have an effect on the calculated peak fuel enthalpy that is dominated by Doppler feedback and Moderator feedback (non-adiabatic calculation). CRDA scenarios starting at higher temperatures and higher powers, where IRM or APRM Setdown scrams are not present, are less limiting with respect to the calculated peak fuel enthalpy. For these scenarios, in addition to Doppler feedback, the increased generation of voids provides an additional, more significant, negative feedback mechanism; so that it is not necessary to credit the scram. See Figures 4-1, 4-2, and 4-3. The moderator feedback is effective where IRM is active and is more effective in reducing the reactivity increase during high power conditions.

Figures 4-1, 4-2 and 4-3 were obtained from BWR Best-Estimate Calculations, as summarized in Reference 11, with the following inputs:

- Realistic feedback due to Doppler, fluid temperature, and voids
- Realistic control blade worths at different exposures
    - Covered static blade worths to limits set by BPWS limits (~1.1%ΔK ~$1.87)
    - Some conservative out-of-sequence cases were included
    - Core loading and shutdown margin limits possible worths
- Different initial fluid temperatures used because feedback mechanisms are different

One of the conclusions derived from Reference 11 is that "responses for calculations in the operating range above ~5% are bounded by startup cases and do not indicate boiling transition."

In summary, the postulated PRNM failure would not adversely affect the plant protection during a CRDA.

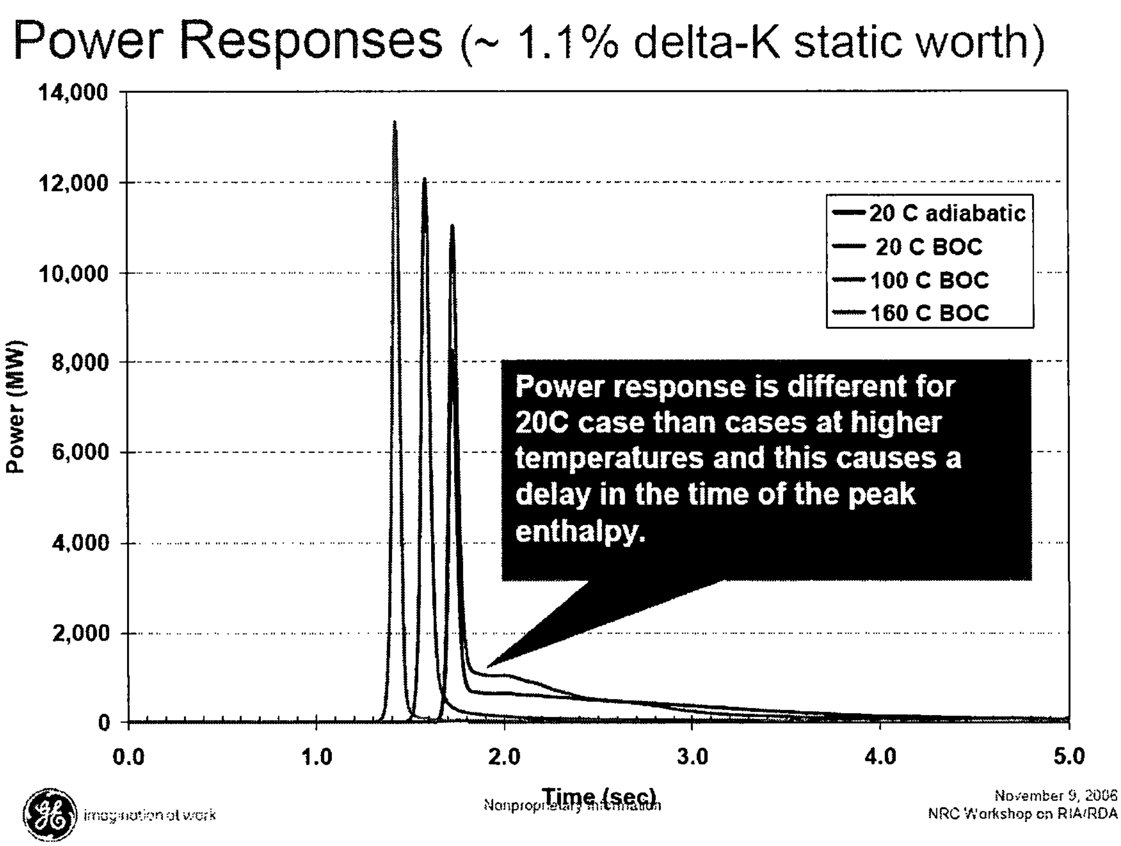

Figure 4-1: Power Response for Realistic CRDA Analysis (from Reference 11)

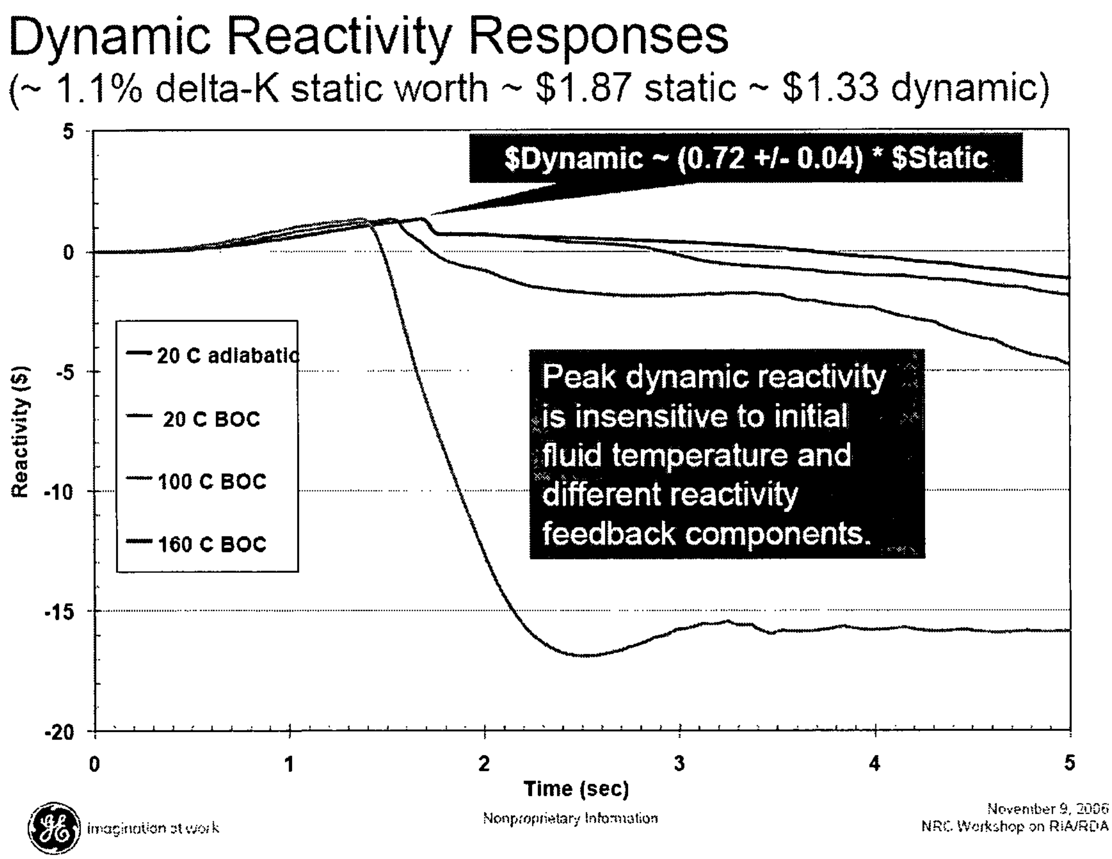

Figure 4-2: Dynamic Reactivity Responses for Realistic CRDA Analysis (from Reference 11)

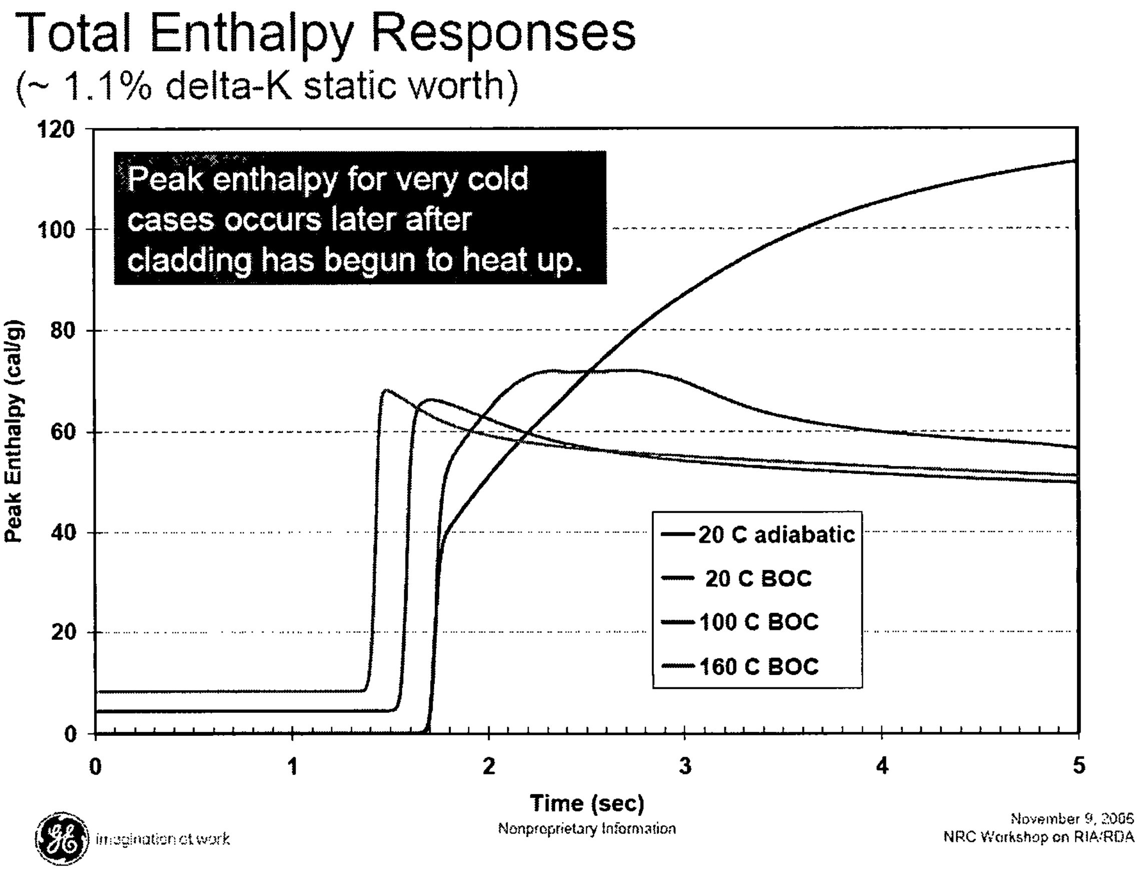

Figure 4-3: Total Enthalpy Responses for Realistic CRDA Analysis (from Reference 11)

In conclusion, based on the evaluation presented in Table 4-1, the proposed upgrade satisfies Reference 3 Acceptance Criteria (1) and (2).

Acceptance Criterion (6)

(6) *For safety systems to satisfy IEEE Std. 603–1991 Clauses 6.2 and 7.2, which are incorporated by reference in 10 CFR 50.55a(h), a safety-related means shall be provided in the control room to implement manual initiation at the division level of the RTS and ESFAS functions. The means provided shall minimize the number of discrete operator manual manipulations and shall depend on operation of a minimum of equipment. If the means is independent and diverse from the safety-related automatically initiated RTS and ESFAS functions, the design meets the system-level actuation criterion in Point 4 of this BTP. If credit is taken for a manual actuation method that meets both the IEEE Std. 603–1991, Clauses 6.2 and 7.2 requirements and a need for a diverse manual backup, then the applicant/licensee should demonstrate that the criteria are satisfied and sufficient diversity exists.*

This criterion requires a safety-related means for manual initiation of the reactor trip system (RTS) and engineered safety features actuation system (ESFAS) functions.

This criterion is not applicable to the PRNM system upgrade. The evaluation performed for Acceptance Criteria (1) and (2) demonstrates that if a CCF occurs in the PRNM system, the plant is able to cope without relying on a manual scram or engineered safety features (ESF) actuation. It is noted that the manual scram and ESF actuation are retained, if needed for other reasons, because they are totally separate from the PRNM system and not affected by the proposed upgrade in any way.

Acceptance Criteria (7) through (9)

(7) *If the D3 assessment reveals a potential for a CCF, then the method for accomplishing the independent and diverse means of actuating the protective safety functions can be accomplished via either an automated system (see Section 3.4, "Use of Automation in Diverse Backup Safety Functions" below), or manual operator actions that meet HFE acceptability criteria (see Section 3.5, "Use of Manual Action in Diverse Backup Safety Functions" below).*

(8) *If the D3 assessment reveals a potential for a CCF, then the method for accomplishing the independent and diverse means of actuating the protective safety functions should meet the following criteria: The independent and diverse means should be:*

- *a) at the division level;*
- *b) initiated from the control room;*
- *c) capable of responding with sufficient time available for the operators to determine the need for protective actions even with malfunctioning indicators, if credited in the D3 coping analysis;*
- *d) appropriate for the event;*
- *e) supported by sufficient instrumentation that indicates:*
    1. *the protective function is needed,*
    2. *the safety-related automated system did not perform the protective function, and*
    3. *the automated backup or manual action is successful in performing the safety function.*

(9) *If the D3 assessment reveals a potential for a CCF, then, in accordance with the augmented quality guidance for the independent and diverse backup system used to cope*

with a CCF, the design of a diverse automated or diverse manual backup actuation system should address how to minimize the potential for a spurious actuation of the protective system caused by the diverse system. Use of design techniques (for example: redundancy, conservative setpoint selection, and use of quality components) to mitigate these concerns is recommended.

These criteria require evaluations of the methods for accomplishing the independent and diverse means of actuating the protective safety function when the D3 analysis reveals the potential for a CCF.

The NUMAC platform is not present in any part of RTS except the PRNMS, and is not present in the ESFAS. Their designs are not affected by the proposed upgrade, and these systems are not vulnerable to the postulated CCF in the PRNMS.

4.3 Additional Diversity Analysis - Undetected Power Oscillations

The OPRM plays an important role in the detection and suppression of power oscillations. The postulated CCF, assumed to result in comprehensive loss of PRNM system functionality, would also disable the OPRM. Although Reference 8 does not include power oscillations among the AOO or DBA, it is appropriate to discuss them. As discussed above, the postulated CCF in the PRNM system results in the system providing valid indications of plant conditions until the transient, at which time they become anomalous. In the case of power oscillations, therefore, PRNM system indications of power and flow must track consistently with other plant indicators as they change to a state point where the potential exists for high growth-rate power oscillations (i.e., the upper left corner of the power/flow map), but somehow fail to provide any protection if large amplitude oscillations begin to occur. Nevertheless, even while maintaining the severity of the postulated CCF, the plant has the ability to cope with it in conjunction with power oscillations.

CGS procedures require immediate action to reduce reactor power or increase core flow in order to mitigate possible high growth rate power oscillations following unanticipated core flow reduction events, such as a two-recirculation pump trip. The operators will know the state point because the status of recirculation pumps is provided independent of PRNM system, flow information is available from the recirculation flow system, and power level information is available from either the electrical power output or a core thermal power calculation. Furthermore, the Recirculation Flow control system, RMCS, and manual scram are unaffected by the CCF. Thus, the plant is able to cope with the CCF because they can determine that defensive steps are necessary and execute those steps.

5. BTP 7-19 DEFENSE IN DEPTH ANALYSIS

5.1 Analysis Approach

Reference 3 Section 1.1 and NUREG/CR-6303, *Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems* (Reference 7) Section 2.2 identify and describe four "echelons of defense," which are "specific applications of the principle of defense-in-depth to the arrangement of instrumentation and control systems attached to a nuclear reactor for the purpose of operating the reactor or shutting it down and cooling it."

The four echelons of defense that are identified, and their applicability to CGS, are as follows.

(1) Control System – consists of (usually) non-safety equipment that is used in the normal operation of a nuclear power plant (NPP) and routinely prevents operations in unsafe regimes of NPP operations. The CGS control system fits this definition. Part of this echelon is the PRNMS, which provides inputs to the RMCS.

(2) Reactor Trip System (RTS) – consists of safety equipment designed to reduce reactivity rapidly in response to an uncontrolled excursion. The CGS safety equipment collectively referred to as the RPS fits this definition. The PRNMS is one of the sensor systems in this echelon. The other equipment, as well as the actuation system, are not affected by the upgrade.

(3) Engineered Safety Features Actuation System (ESFAS) – consists of safety equipment that removes heat or otherwise assists in maintaining the integrity of the three physical barriers to radioactive release. The CGS ESFAS complies with this definition. The PRNMS does not interface with this echelon.

(4) Monitoring and Indicators – consists of sensors, displays, data communication systems, and manual controls required by operators to respond to NPP operating events. The CGS display and other instrumentation systems fit this definition. The PRNM system provides inputs to this echelon (e.g., annunciators) but does not receive any signals from this echelon.

It is of particular importance to ensure that no single CCF will disable more than one echelon.

5.2 Evaluation

Acceptance Criterion (3)

(3) When a failure of a common element or signal source shared by the control system and RTS is postulated and the CCF results in a plant response that requires reactor trip and also impairs the trip function, then diverse means that are not subject to or failed by the postulated failure should be provided to perform the RTS function. The diverse means should assure that the plant response calculated using realistic assumptions analyses does not result in radiation release exceeding 10 percent of the applicable siting dose guideline values or violation of the integrity of the primary coolant pressure boundary.

This criterion requires an evaluation of potential interaction between the Control System and RTS echelons when a postulated CCF results in a plant response that requires a reactor trip and also impairs the trip function. The PRNM system is not used for automatic control of plant operations, except for providing rod block signals. Therefore, if the postulated CCF occurs, it will not result in a plant response that requires a reactor trip. Therefore, the type of CCF described in this criterion cannot occur in the upgrade system. Acceptance Criterion (3) is satisfied.

Acceptance Criterion (4)

(4) When a failure of a common element or signal source shared by the control system and ESFAS is postulated and the CCF results in a plant response that requires engineered safety features (ESF) and also impairs the ESF function, then diverse means that are not subject to or failed by the postulated failure should be provided to perform the ESF function. The diverse means should assure that the plant response calculated using realistic assumptions analyses does not result in radiation release exceeding 10 percent of the applicable siting dose guideline values or violation of the integrity of the primary coolant pressure boundary

This criterion requires an evaluation of potential interactions between the Control System and ESFAS echelons when a postulated CCF results in a plant response that requires an ESF response and also impairs ESF function. The PRNM system is not used for automatic control of plant operations, so if the postulated CCF occurs, it will not result in a plant response that requires an ESF response. Furthermore, neither the existing nor replacement PRNMS interface with the ESFAS. Therefore, the type of CCF described in this criterion cannot occur in the upgrade system. Acceptance Criterion (4) is satisfied.

Acceptance Criterion (5)

(5) No failure of monitoring or display systems should influence the functioning of the RTS or ESFAS. If a plant monitoring system failure induces operators to attempt to operate the plant outside safety limits or in violation of the limiting conditions of operation, the analysis should demonstrate that such operator-induced transients will be compensated by protection system function.

This criterion requires that a failure in the monitoring and display echelon will not adversely affect the RTS or ESFAS echelons. The PRNM system does not rely on or receive any input from the monitoring and display echelon, and therefore, a failure in the monitoring and display systems will not propagate to PRNMS. If the failure in the monitoring and display system results in an operator-induced transient, the automatic protective functions of PRNM system are available for compensation. Acceptance Criterion (5) is satisfied.

6. SUMMARY & CONCLUSIONS

The PRNM system upgrade was evaluated using the Acceptance Criteria identified in NRC BTP 7-19, Revision 6. This report provides an assessment of diversity and defense-in-depth, using the original LTR (Reference 1). Additionally, this report provides a detailed D3 analysis based on a postulated worst-case common-cause failure in the PRNM system programmable entities, and directly addresses all criteria of BTP 7-19. The evaluations demonstrate that the plant has the diversity and defense-in-depth to cope with any potential CCF in the programmable entities in the upgrade system.

7. REFERENCES

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4. NRC letter to Mr. David W. Reigel, NUMAC Project Manager, Acceptance of Licensing Topical Report NEDC- 32410-P, Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function (TAC No. M90616), September 5, 1995; and

   NRC letter to Mr. David W. Reigel, NUMAC Project Manager, Licensing Topical Report NEDC- 32410-P, Supplement 1, Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function (TAC No. M95746), August 15, 1997.

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