Entergy

Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

Michael A. Krupa
Director, Extended Power Uprate
Grand Gulf Nuclear Station
Tel. (601) 437-6684

Attachments 1 and 2 contain PROPRIETARY information.

GNRO-2010/00051

July 29, 2010

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: Responses to NRC Request for Additional Information Pertaining to License Amendment Request for Power Range Neutron Monitoring System (TAC No. ME2531)

Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

REFERENCES:

1. Entergy Operations, Inc. letter to the NRC (GNRO-2009/00054), *License Amendment Request – Power Range Neutron Monitoring System Upgrade*, November 3, 2009 (ADAMS Accession No. ML093140463)

2. NRC letter to Entergy Operations, Inc. (GNRI-2010/00067), *Grand Gulf Nuclear Station, Unit 1 - Request for Additional Information Re: Power Range Neutron Monitoring System (TAC No. ME2531)*, May 4, 2010 (ADAMS Accession No. ML101190125)

3. Entergy Operations, Inc. letter to the NRC (GNRO-2010/00040), *Responses to NRC Requests for Additional Information Pertaining to License Amendment Request for Power Range Neutron Monitoring System (TAC No. ME2531)*, June 3, 2010

4. Entergy Operations, Inc. letter to the NRC (GNRO-2010/00045), *Responses to NRC Requests for Additional Information Pertaining to License Amendment Request for Power Range Neutron Monitoring System (TAC No. ME2531)*, June 18, 2010

Dear Sir or Madam:

In Reference 1, Entergy Operations, Inc. (Entergy) submitted to the NRC a license amendment request (LAR), which proposes to revise the Grand Gulf Nuclear Station (GGNS) Technical Specifications to reflect installation of the digital General Electric-Hitachi (GEH) Nuclear Measurement Analysis and Control (NUMAC) Power Range Neutron Monitoring System.

When Attachments 1 and 2 are removed from this letter, the entire document is NON-PROPRIETARY.

A001
NRR

In Reference 2, the NRC staff transmitted to Entergy a request for additional information (RAI) needed to support their review and approval of the LAR. Entergy submitted responses to several of the RAIs via Reference 3; however, responses to RAI Nos. 1, 2, and 3 were not provided at that time. Entergy committed to provide responses to those RAIs on or before July 29, 2010 in Reference 4. These responses are contained in Attachments 1 through 4.

GEH considers certain information contained in Attachments 1 and 2 to be proprietary and, therefore, requests it be withheld from public disclosure in accordance with 10 CFR 2.390. GEH, as the owner of the proprietary information, has executed the affidavit contained in Attachment 5. This affidavit identifies that the enclosed proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. The proprietary information was provided to GGNS in a GEH transmittal that is referenced by the affidavit. The proprietary information has been faithfully reproduced in Attachments 1 and 2 such that the affidavit remains applicable. GEH hereby requests that the enclosed proprietary information be withheld from public disclosure in accordance with the provisions of 10 CFR 2.390 and 9.17. Non-proprietary versions of Attachments 1 and 2 are provided in Attachments 3 and 4, respectively.

The No Significance Hazards Determination and the Environmental Consideration provided in Reference 1 are not impacted by these responses.

Also, in Reference 7, Entergy responded to RAI No. 7. In its response, Entergy committed to provide the requested human factors review information on or before June 30, 2011. Since submitting Reference 7, Entergy has reassessed the actions associated with developing the information and has determined that it can be provided sooner than originally committed. Therefore, Entergy revises the previous commitment and now commits to provide the human factors information requested in RAI No. 7 on or before January 17, 2011.

This letter contains one revised commitment, which is identified in Attachment 6.

If you have any questions or require additional information, please contact Mr. Guy Davant at (601) 368-5756.

I declare under penalty of perjury that the foregoing is true and correct; executed on July 29, 2010.

Sincerely,

M. A. Krupa

MAK/ghd

Attachments:

1. Responses to NRC Request for Additional Information Pertaining to License Amendment Request – Power Range Neutron Monitoring System Upgrade (Proprietary Version)
2. Response to RAI No. 1 - GGNS PRNM System Platform Description (Proprietary Version)
3. Responses to NRC Request for Additional Information Pertaining to License Amendment Request – Power Range Neutron Monitoring System Upgrade (Non-Proprietary Version)
4. Response to RAI No. 1 - GGNS PRNM System Platform Description (Non-Proprietary Version)
5. Affidavit Supporting Request to Withhold Information from Public Disclosure
6. List of Regulatory Commitments

cc:

Mr. Elmo E. Collins, Jr.
Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
612 East Lamar Blvd., Suite 400
Arlington, TX 76011-4005

U. S. Nuclear Regulatory Commission
ATTN: Mr. C. F. Lyon, NRR/DORL (w/2)
Mail Stop OWFN/8 B1
11555 Rockville Pike
Rockville, MD 20852-2378

NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

ATTACHMENT 3

GNRO-2010/00051

RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION PERTAINING TO LICENSE AMENDMENT REQUEST POWER RANGE NEUTRON MONITORING SYSTEM UPGRADE

(NON-PROPRIETARY VERSION)

This is a non-proprietary version of Attachment 1 from which the proprietary information has been removed. The proprietary portions that have been removed are indicated by double square brackets as shown here: [[ ]].

RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION PERTAINING TO LICENSE AMENDMENT REQUEST POWER RANGE NEUTRON MONITORING SYSTEM UPGRADE

By application dated November 3, 2009, Entergy Operations, Inc. (Entergy) submitted to the NRC a license amendment request (LAR) for the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications to reflect installation of the digital General Electric - Hitachi (GEH) Nuclear Management Analysis and Control (NUMAC) Power Range Neutron Monitoring (PRNM) System (Reference 1). In Reference 2, the NRC Staff transmitted to Entergy a request for additional information (RAI) needed to support their review and approval of the PRNM System LAR. Entergy submitted responses to several of the RAIs via Reference 3; however, responses to RAI Nos. 1, 2, and 3 were not provided at that time. Entergy committed to provide responses to those RAIs on or before July 29, 2010 in Reference 4. These responses are provided below.

NRC RAI No. 1

Please identify the changes to the GEH NUMAC PRNM System platform from that defined and approved within GE LTR NEDC-32410P-A. For example, the identified changes should include those to hardware, programmable devices, software. Applicable development processes, and the like, that will be reflected within the GGNS PRNM System upgrade. When considering the software development processes for the platform, the response should address changes (from that previously approved for the GE LTR NEDC-32410P-A) to the applicable documentation that is identified under Section B.2 of Standard Review Plan (SRP), NUREG-0800, Branch Technical Position 7-14, "Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems," Revision 5, March 2007 (ADAMS Accession No. ML070670183).

Response

The GGNS PRNM System is being designed in accordance with the requirements of GE Nuclear Energy Licensing Topical Report (LTR) NEDC-32410P-A, *Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function,* Volumes 1 and 2, including Supplement 1, referred to herein collectively as the PRNM LTR (Reference 5). The NRC approved the PRNM LTR in safety evaluation reports (SERs) dated September 5, 1995 (for Vols. 1 and 2) and August 15, 1997 (for Supplement 1) (Reference 6).

Attachment 2 of this letter contains a thorough, detailed description of the GGNS PRNM System platform, including changes made to the platform since approval of the PRNM LTR. Changes result from equipment obsolescence, new and updated industry standards, improved manufacturing techniques, updated software, etc. These changes do not adversely impact or result in any deviation in operation or functionality of the PRNM System as described in the PRNM LTR.

For clarity, the platform description provided in Attachment 2 is presented in five parts:

(1) Hardware Changes;

(2) Software Changes;

(3) Software Process Changes;

(4) Programmable Logic Device (PLD) Process; and

(5) Regulatory Compliance.

NRC RAI No. 2

NUMAC PRNM LTR Section 4.4.1.11 requires the Utility Action to "identity the specific requirements applicable to the plant, confirm that any clarifications included in NEDC-32410P-A apply to the plant, and document the specific requirements that the replacement PRNM is intended to meet for the plant." However, the entry for 4.4.1.11 in Attachment 2 of the license amendment does not document specific GGNS PRNM requirements. For example, no discussion of response time requirements could be identified. Therefore, to support the NRC staff's review using current criteria, please identify the GGNS specific requirements applicable to the GEH NUMAC PRNM System application (e.g., application software). When considering the requirements documentation, please address the related documentation that is identified under Section B.2 of SRP, NUREG-0800, Branch Technical Position 7-14, Revision 5.

Response

The plant-specific PRNM System platform is described in the PRNM LTR with changes identified and described in the response to RAI No.1, above. Part 3 of the response to RAI No. 1 discusses documentation identified under Section B.2 of Branch Technical Position (BTP) 7-14 (Reference 7). In addition to the information provided in the response to RAI No. 1, Entergy provided in its response to RAI No. 6 (Reference 3) additions to and deletions from the PRNM LTR descriptions that reflect the GGNS application. Additional information pertaining to GGNS-specific requirements is presented below.

Tables 2-1 and 2-2, below, present the plant-specific requirements the PRNM System must meet. Specifically, Table 2-1 identifies:

(1) The GGNS-specific regulatory requirements, codes, and standards applicable to the PRNM System; and

(2) The applicable PRNM LTR sections that address the requirements.

TABLE 2-1

PLANT-SPECIFIC REGULATORY REQUIREMENTS, CODES, AND STANDARDS

REQUIREMENT	TITLE	PRNM LTR SECTION
IEEE 279-1971	Criteria for Protection Systems for Nuclear Power Generating Stations	4.4.1.1
IEEE 323-1971	Qualifying Class 1E Equipment for Nuclear Power Generating Stations	4.4.2.2.1.2 (Adheres to IEEE 323-1983.)
IEEE 336-1971	Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations	4.4.1.8
IEEE 338-1971	Criteria for Periodic Testing of Nuclear Power Generating Station Safety Systems	4.4.1.8
IEEE 344-1971	Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations	4.4.2.3.2
IEEE 379-1972	Application of the Single-Failure Criterion to Nuclear Power Generating Station Class 1E Systems	4.4.1.8
RG 1.22 (2/1972)	Periodic Testing of Protection System Actuation Functions	4.4.1.7
RG 1.29 (9/1978)	Seismic Design Classification	4.4.1.7
RG 1.47 (5/1973)	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	4.4.1.7
RG 1.53 (6/1973)	Application of Single-Failure Criterion to Nuclear Power Plants	4.4.1.7
RG 1.63 (10/1973)	Electric Penetrations Assemblies in Containment Structures for Nuclear Power Plants	4.4.1.7
RG 1.75 (1/1975)	Physical Independence of Electric Systems	4.4.1.6
RG 1.89 (11/1974)	EQ of Certain Electric Equipment Important to Safety for Nuclear Power Plants	4.4.2.2.1.2 (Adheres to IEEE 323-1983.)
RG 1.97 (12/1980)	Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants	4.4.1.2

Table 2-2 identifies:

(1) Plant-specific system performance parameters and associated Reactor Protection System (RPS) requirements of the current Average Power Range Monitor (APRM) subsystem applicable to the PRNM System;

(2) The associated PRNM System performance specification; and

(3) The associated PRNM LTR section that documents the performance specification.

TABLE 2-2

PLANT-SPECIFIC SYSTEM PERFORMANCE REQUIREMENTS

RPS Function	Performance Parameter	RPS Requirement	PRNM System Performance	PRNM LTR Section
APRM Neutron Flux – High	Response Time	≤ 0.09 second	[[ ]]	3.3.2
APRM Flow Biased Simulated Thermal Power – High	Response Time	≤ 0.09 second	[[ ]]	3.3.2

Additional Discussion Regarding RAI No. 2

During a telephone call on June 1, 2010, the NRC Staff and Entergy discussed the scope of RAI No. 2 (documented in Reference 8). In that call, the NRC discussed their intent to review the PRNM System LAR against IEEE Standard (Std.) 603-1991, *IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations* (Reference 9).

Entergy has reviewed the application of IEEE standards to the PRNM System design as documented in the PRNM LTR and the NRC's SER, the application rules for IEEE Std. 279-1971 (Reference 10) and IEEE Std. 603-1991 specified in 10 CFR 50.55a(h)(2) (Reference 11), the Statements of Consideration for 10 CFR 50.55a(h)(2) (Reference 12), and the guidance for review provided in SRP Section 7.1, "Instrumentation and Controls - Introduction," and its appendices (Reference 13). Each is discussed below.

- PRNM LTR and SER

  Section 4.4 of the PRNM LTR identifies the regulatory requirements applied to the PRNM System, IEEE Std. 279-1971 being the primary standard used to design the system. Although this was the case, the NRC Staff accepted GEH's review of the design against a previous version of IEEE Std. 603, as documented in Section 3.2 of the SER, which stated in part:

  > "The correlation of the software development process to the criteria of IEEE Std. 603-1980, 'IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,' (which is endorsed in RG 1.153, 'Criteria for Power, Instrumentation, and Control Portions of Safety Systems,') is also presented in Appendix A of NEDC-32410. The correlation is consistent with staff criteria and is acceptable. The NRC staff finds the software design process acceptable."

- 10 CFR 50.55a(h)(2)

  10 CFR 50.55a(h)(2), *Protection systems*, states:

  > "For nuclear power plants with construction permits issued after January 1, 1971, but before May 13, 1999, protection systems must meet the requirements stated in either IEEE Std. 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," or in IEEE Std. 603-1991, "Criteria for Safety Systems for Nuclear Power Generating Stations," and the correction sheet dated January 30, 1995. For nuclear power plants with construction permits issued before January 1, 1971, protection systems must be consistent with their licensing basis or may meet the requirements of IEEE Std. 603-1991 and the correction sheet dated January 30, 1995."

  The PRNM System is a "protection system" as defined in both IEEE 279-1971 and IEEE 603-1991; therefore, this regulation is applicable.

  GGNS received its construction permit on September 4, 1974. As allowed by 10 CFR 50.55a(h)(2), Entergy has elected to comply with IEEE 279-1971.

- Statements of Consideration for 10 CFR 50.55a(h)(2)

  In Federal Register Notice 63FR20136 dated April 23, 1998, which promulgated 10 CFR 50.55a(h)(2), the NRC discussed application of IEEE 279 and IEEE 603 to then-current licensees, as follows:

  > "Current licensees may continue to meet the requirements stated in the edition or revision of IEEE Std. 279 in effect on the formal date of their application for a construction permit or may, at their option, use IEEE Std. 603–1991, provided they comply with all applicable requirements for making changes to their licensing basis."

  The NRC then stated that "system-level replacements of protection systems and addition of new safety systems in operating nuclear power plants initiated on or after January 1, 1999, would be required to meet the requirements in IEEE Std. 603–1991." However, the PRNM System upgrade to the current APRM subsystem does not meet this definition, as specifically discussed by the NRC in the notice, as follows:

  > "A 'system' is defined as a combination of two or more interrelated components that perform a specific safety function. The protection systems are listed in the plant's FSAR. For example, 'neutron monitoring system' is a protection system. The upgrade of the average power range monitor (APRM) portion of the neutron monitoring system to add the ability to detect and suppress potential boiling water reactor (BWR) instability may meet IEEE Std. 279 because the modification only replaces the APRM signal processing components, output relays, recirculation flow transmitters, and operator displays."

- SRP Section 7.1 and Appendices

  Section 7.1 and associated Appendices 7.1-A through 7.1-D restate the application rules of 10 CFR 50.55a(h)(2), thereby recognizing the option of a licensee who received a construction permit after January 1, 1971, but before May 13, 1999, to comply with either

IEEE Std. 279-1971 or IEEE Std. 603-1991 for protection systems. Since Entergy has elected to comply with IEEE Std. 279-1971, only guidance for evaluating compliance with this standard is required when reviewing the PRNM System LAR. Therefore, Appendices 7.1-C, "Guidance for Evaluation of Conformance to IEEE Std. 603," and 7.1-D, "Guidance for Evaluation of Conformance to IEEE Std. 7-4.3.2," are not applicable. Please note that these two appendices refer the reviewer to Appendix 7.1-B, "Guidance for Evaluation of Conformance to IEEE Std. 279," for guidance when applying the requirements of IEEE Std. 279-1971.

IEEE Std. 7-4.3.2, *Application Criteria for Programmable Digital Computer Systems of Nuclear Power Generating Stations* (Reference 14), specifies additional computer-specific requirements to supplement the criteria and requirements of IEEE Std. 603. The PRNM LTR provides a correlation of the PRNM design process to IEEE Std. 7-4.3.2-1993 in Table A.2 of Appendix A. An updated correlation to IEEE Std. 7-4.3.2-2003 is provided in the response to RAI No. 1 (see Table 1-11 of Attachment 2).

Conclusion

Based on the above information, Entergy believes that additional reviews of the PRNM System LAR against IEEE Std. 603 are not required.

NRC RAI No. 3

Please describe how a software common-cause failure of the GGNS PRNM System upgrade application is addressed or coped with, such that upon its failure GGNS remains within its design basis for all design-basis accidents and anticipated operational occurrences.

Response

The existing APRM subsystem provides a single-sensor input to 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 sensor inputs into RPS (e.g., reactor dome pressure) are diverse from the PRNM System and do not utilize the NUMAC platform. Therefore, they are not subject to the same common-cause failures.

An in-depth analysis of common-cause software-related failures for the PRNM System was previously performed by GEH, as documented in the PRNM LTR (Reference 5) and approved by the NRC in their SER approving the LTR (Reference 6). Relevant information from each document is presented below.

PRNM LTR

Section 6.4 of the PRNM LTR discusses a qualitative assessment of the effect of common-mode and common-cause failures on the overall plant (defense-in-depth), specifically addressing such failures of the APRM and Oscillation Power Range Monitor (OPRM) functions in Sections 6.4.1 and 6.4.2, respectively. For the APRM functions, Section 6.4.1 references analyses documented in GEH NEDC-30851P-A, *Technical Specification Improvement Analysis for BWR Reactor Protection System,* March 1988 (Reference 15),

which employs EPRI Report No. NP-2230, Part 3, *ATWS: Frequency of Anticipated Transients.*[1]

Section 6.4.1 states in part:

[[

]]

Table F-1 of NEDC-30851P-A is reproduced below.

TABLE F-1

SENSOR DIVERSITY FOR INITIATING EVENTS

[[								
					(See Note 1)			
(See Note 2)								
(See Note 2)								
(See Note 3)								
								]]

[1] The NRC approved NEDC-30851P-A in a letter to the BWR Owners' Group dated January 24, 1988 (Reference 16).

TABLE F-1

SENSOR DIVERSITY FOR INITIATING EVENTS
(continued)

[[					(See Note 1)			
								]]

Note 1: The GGNS design initiates a Turbine Stop Valve Closure reactor scram on low trip fluid pressure indicative of less than 90% full open, rather than actual valve position. Therefore, the Turbine Control Valve scram and the Turbine Stop Valve scram are not diverse RPS functions for GGNS. However, this design does not adversely impact the conclusions of NEDC-30851P-A as applied to GGNS since there exist other diverse RPS functions for the events that utilize these scrams, as identified in the table.

Note 2: GGNS 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.

Note 3: GGNS is not specifically analyzed for this event. GEH 10 CFR Part 21 Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," documents that reactor vessel level swell resulting from a Pressure Regulator Failure – Maximum Demand (Open) may not be sufficient to cause a high reactor water level scram. Therefore, this event is bounded by the Pressure Regulator Failure (Primary Pressure Decrease) (MSIV Closure Trip) event.

Regarding the OPRM function, Section 6.4.2 states:

[[

]]

Section 6.5 of the PRNM LTR documented the following conclusions:

[[

]]

Please refer to the PRNM LTR for more information.

NRC Safety Evaluation Report

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 their SER. 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 common cause failures in the software. The staff finds the above features to address malfunctions to be acceptable."

Application to GGNS

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.

Entergy confirmed these three items in Section 6.6 of Attachment 2 of the PRNM System LAR (Reference 1). Additional information supporting this confirmation is provided below.

(1) Table 3-1, below, lists both sets of events identified in Appendices F and G of NEDC-30851P-A (Reference 15) and identifies the applicable section the GGNS UFSAR in which the event is discussed.

(2) The GGNS-specific PRNM System configuration is described in the response to RAI No. 1, which documents it is within the limits described in the PRNM LTR.

(3) As stated in Section 6.6 of Attachment 2 of the PRNM System LAR (Reference 1), the requirements of 10 CFR 50.59 apply to the PRNMS modification in accordance with applicable plant procedures.

TABLE 3-1

CROSS-REFERENCE OF NEDC-30851P-A EVENTS TO GGNS UFSAR

IDENTIFIED EVENT	UFSAR SECTION
Appendix F – Transient/Accidents Analyses	
Main Steam Isolation Valve (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)	See Note 2
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
Loss of Instrument Air	15.2.10
Recirculation Flow Control Failure (Increase Flow)	15.4.5
Recirculation Flow Control Failure (Decreasing Flow)	15.3.2

TABLE 3-1

CROSS-REFERENCE OF NEDC-30851P-A EVENTS TO GGNS UFSAR
(continued)

IDENTIFIED EVENT	UFSAR SECTION
Inadvertent Opening of One Safety/Relief Valve	15.1.4
Inadvertent 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
High Flux due to Rod Withdrawal at Startup	15.4.1
Inadvertent Insertion of Control Rods	15.4.3
Detected Fault in RPS	See Note 3
Inadvertent startup of HPCI/HPCS	15.5.1
Scram due to Plant Occurrences (Manual Scram)	See Note 3
Spurious Trip via Instrumentation, RPS Fault	See Note 3
Manual Scram – No Out-of-Tolerance Condition	See Note 3

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

Note 2: GGNS is not specifically analyzed for this event. GEH 10 CFR Part 21 Communication SC05-03, "Potential to Exceed Low Pressure Technical Specification Safety Limit," documents that reactor vessel level swell resulting from a Pressure Regulator Failure – Maximum Demand (Open) may not be sufficient to cause a high reactor water level scram. Therefore, this event is bounded by the Pressure Regulator Failure (Primary Pressure Decrease) (MSIV Closure Trip) event.

Note 3: This event is considered an anticipated operational occurrence (AOO) within the design and licensing bases of the unit and does not encroach upon a safety limit. As an AOO, it is not specifically identified in the UFSAR.

REFERENCES

1. Entergy Operations, Inc. letter to the NRC (GNRO-2009/00054), *License Amendment Request – Power Range Neutron Monitoring System Upgrade*, November 3, 2009

2. NRC letter to Entergy Operations, Inc. (GNRI-2010/00067), *Grand Gulf Nuclear Station, Unit 1 - Request for Additional Information Re: Power Range Neutron Monitoring System (TAC No. ME2531)*, May 4, 2010

3. Entergy Operations, Inc. letter to the NRC (GNRO-2010/00040), *Responses to NRC Requests for Additional Information Pertaining to License Amendment Request for Power Range Neutron Monitoring System (TAC No. ME2531)*, June 3, 2010

4. Entergy Operations, Inc. letter to the NRC (GNRO-2010/00045), *Responses to NRC Requests for Additional Information Pertaining to License Amendment Request for Power Range Neutron Monitoring System (TAC No. ME2531)*, June 18, 2010

5. GE Nuclear Energy LTR NEDC-32410P-A Volume 1 and NEDC-32410P-A Volume 2 -- Appendices, *Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function*, dated October 1995 (ADAMS Accession No. ML9605290009); and

   GE Nuclear Energy LTR NEDC-32410P-A Supplement 1, *Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function*, dated November 1997 (ADAMS Accession No. ML9806120242)

6. 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

7. NRC Branch Technical Position (BTP) 7-14, *Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems*, of NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*

8. NRC meeting minutes *Summary of Meeting on May 25, 2010, to Discuss the Licensee's Response to the NRC Staff's Request of Additional Information dated May 4, 2010, Regarding Installation of a digital Power Range Neutron Monitoring System (TAC No. ME2531)*, June 9, 2010, (Accession No. ML101520269)

9. Institute of Electrical and Electronics Engineers Standard 603-1991, *IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations*

10. Institute of Electrical and Electronics Engineers Standard 279-1971, *IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations*

11. Institute of Electrical and Electronics Engineers Standard 7-4.3.2, *IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations*

12. Paragraph (h)(2) of 10 CFR 50.55a, *Codes and Standards*

13. Federal Register Notice 63FR20136, April 23, 1998

14. Section 7.1, *Instrumentation and Controls – Introduction*, of NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*

15. GE Nuclear Energy NEDC-30851P-A, *Technical Specification Improvement Analysis for BWR Reactor Protection System*, March 1988

16. NRC letter to Mr. Robert F. Janacek, Chairman, BWR Owners' Group, *General Electric Company (GE) Topical Reports NEDC-30844, "BWR Owners Group Response to NRC Generic Letter 83-28," and NEDC-30831P [sic], "Technical Specifications Improvement Analysis for BWR RPS – Use for Solid State Plants,"* January 24, 1988

ATTACHMENT 4

GNRO-2010/00051

RESPONSE TO RAI No. 1
GGNS PRNM SYSTEM PLATFORM DESCRIPTION

(NON-PROPRIETARY VERSION)

This is a non-proprietary version of Attachment 2 from which the proprietary information has been removed. The proprietary portions that have been removed are indicated by double square brackets as shown here: [[ ]].

RESPONSE TO RAI No. 1
GGNS PRNM SYSTEM PLATFORM DESCRIPTION

NRC RAI No. 1

Please identify the changes to the GEH NUMAC PRNM System platform from that defined and approved within GE LTR NEDC-32410P-A. For example, the identified changes should include those to hardware, programmable devices, software. Applicable development processes, and the like, that will be reflected within the GGNS PRNM System upgrade. When considering the software development processes for the platform, the response should address changes (from that previously approved for the GE LTR NEDC-32410P-A) to the applicable documentation that is identified under Section B.2 of Standard Review Plan (SRP), NUREG-0800, Branch Technical Position 7-14, "Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems," Revision 5, March 2007 (ADAMS Accession No. ML070670183).

Response

The response is prepared in five parts:

Part 1 provides a discussion of the hardware changes that have been made to the Power Range Neutron Monitor (PRNM) platform by comparing the Grand Gulf PRNM to the Hatch PRNM, which is identical to the platform described in the PRNM LTR. Tables 1-1, 1-2, and 1-3 provide detailed comparisons of the hardware modules used in each application. Table 1-4 provides the detailed change history for all hardware modules.

Part 2 provides a discussion of the software changes that have been made to the PRNM platform, including a description of the overall software change process. Table 1-5 provides the detailed change history for all affected source code modules leading up to the Grand Gulf application.

Part 3 provides a discussion of the changes that have been made to the approved software development process, including changes to the Nuclear Measurement Analysis Control (NUMAC) software plan documents and compliance with Branch Technical Position (BTP) 7-14. Table 1-6 provides the detailed revision history for each of the NUMAC software plan documents. Table 1-7 provides a correlation of the NUMAC PRNM design process to section B.2 of BTP 7-14.

Part 4 provides a discussion of the required BWR 6 voter logic change and the process for making changes to programmable logic devices. Table 1-8 provides a mapping of this process to the approved NUMAC software process.

Part 5 provides a discussion of PRNM compliance with regulatory changes that have occurred since the PRNM LTR was first reviewed and approved. Table 1-9 provides an evaluation of the PRNM against the current applicable Regulatory Guides identified in Table 7-1 of the Standard Review Plan (SRP) NUREG-0800. Table 1-10 provides a correlation of the NUMAC PRNM design process to Regulatory Guide (RG) 1.152 Revision 2 and Table 1-11 provides a correlation of the NUMAC PRNM design process to IEEE 7-4.3.2-2003.

Part 1: NUMAC PRNM Platform Hardware Changes

The first PRNM system installed in the United States was installed at Hatch in 1997. The PRNM platform at Hatch is identical to the platform described in PRNM Licensing Topical Report (LTR) NEDC-32410P-A, and therefore provides a basis for comparison to the platform that was originally reviewed and approved by NRC. Tables 1-1, 1-2, and 1-3 show the differences in the NUMAC platform between the initial U.S. application at Hatch in 1997 and the Grand Gulf PRNM application by comparing the part numbers of the hardware modules used in the Hatch application to the part numbers of the hardware modules used in the Grand Gulf application. Table 1-4 summarizes all the changes to the hardware modules by parts list revision since the initial U.S. application at Hatch. Regardless of any hardware changes that have occurred since the original application, if the part number used for Grand Gulf is the same part number that was used for Hatch then the part is fully interchangeable with respect to form, fit and function in accordance with GEH engineering operating procedures. The following paragraphs provide details of the significant hardware platform changes.

Universal Front Panel

[[

]]

Application-Specific Printed Wiring Board (PWB) Backplane

[[

]]

Universal Chassis Mechanical Assembly

[[

]]

New Low Voltage Power Supply (LVPS) Mounting Brackets

[[

]]

GEDAC Communication/Memory Module

[[

]]

Relay Logic Card

[[

]]

Voter Logic Programmable Logic Device (PLD)

[[

]]

Some of the differences between the initial U.S. application at Hatch in 1997 and the Grand Gulf PRNM application described above have the potential to affect the seismic and environmental qualification of the system. As a result, the new APRM chassis and PCI

chassis designs will be tested to demonstrate compliance with all applicable seismic, environmental, and electromagnetic compatibility (EMC) qualification requirements. Entergy committed to provide this information in a letter to the NRC dated June 3, 2010.

Part 2: NUMAC PRNM Platform Software Changes

Table 1-5 identifies changes made to the safety-related generic APRM/OPRM firmware since the original design that represent the starting point for the Grand Gulf PRNM software development effort. The table lists the files containing revised firmware and a description of the changes. This table does not include changes made to the data files that are changed for each new plant application. These changes have been made in accordance with the NUMAC Verification and Validation (V&V) process and the NUMAC configuration management process that were previously reviewed and approved by the NRC, as stated in section 3.2 of the safety evaluation report of NEDC-32410P-A. The following is a synopsis of the APRM/OPRM software evolution process:

Design Inputs

[[

]]

Firmware Control

[[

]]

Firmware History

[[

]]

Firmware Testing

[[

]]

Future Application

[[

]]

Summary

[[

]]

Part 3: NUMAC PRNM Platform Software Development Process Changes

[[

]] Section 3.2 of the safety evaluation report of LTR NEDC-32410P-A states that the standard NUMAC software development process defined by these plans and implemented for PRNM has been reviewed and accepted by NRC. Consistent with the commitment that was made by GE to NRC as documented in section 3.2 of the safety evaluation report of LTR NEDC-32410P-A, the NUMAC software development plans were issued as formally controlled corporate documents. Since the NRC first reviewed and approved the NUMAC software development plans, several changes have been made to these documents. These document changes were made in accordance with GE procedures and in accordance with the required engineering and quality assurance reviews as was committed to NRC at the time NEDC-32410P-A and these NUMAC software development plans were first reviewed and approved. The changes that have been made to these documents do not in any way alter the fundamental software life cycle process that was originally reviewed and approved by NRC. Table 1-6 summarizes the revision history of the NUMAC software plans since they were first reviewed and approved by NRC. Table 1-7 shows the correlation of the NUMAC design process to the requirements of BTP 7-14 Revision 5.

NUMAC Software Plans Revision History

[[

]]

BTP 7-14 Compliance

The primary NRC guideline available at the time the NUMAC design processes were developed was the NRC RG 1.152 Rev. 0 (1985), primarily endorsing ANSI/IEEE 7-4.3.2-1982. IEEE 7-4.3.2-1993 was issued prior to completion of the original PRNM design, but was not endorsed by the NRC until 1996 (via RG 1.152 Rev. 1). Evaluation of the NUMAC design process against both of those guides is included in NEDC-32410P-A, Appendix A. In addition, NEDC-32410P-A, Supplement 1, Appendix A, includes an evaluation of the process to ANSI NQA2, Part 2.7. A general description of the design process applied to the NUMAC PRNM is included in NEDC-32410P-A, Chapter 9. Finally, Appendix C in NEDC-32410P-A includes a comparison of the NUMAC PRNM equipment with NUMAC equipment previously designed and reviewed by the NRC.

Because the original PRNM design and NRC review of the NUMAC PRNM LTR, the NRC has issued BTP 7-14, Revision 5. This BTP and most of the US NRC RGs listed therein were not issued at the time of the original design of the NUMAC PRNM equipment. BTP 7-14 guidance is intended to address complete digital systems in a plant, including full Reactor Trip Systems and Engineered Safety Features Systems. [[

]] Extensive field experience of

NUMAC equipment, including the PRNM, demonstrates that the design process applied for the NUMAC equipment, including PRNM, provides a fully adequate digital design for the NUMAC applications.

Part 4: Discussion of BWR 6 Voter Logic PLD Development Process

The PRNM voting logic is performed in non-volatile programmable logic devices (PLDs) that are part of the Two-Out-Of-Four Logic Module. The voting logic has not changed since the initial U.S. application at Hatch in 1997. However, for the BWR 6 platform, a change is required to implement the voting logic that was previously reviewed and approved by US NRC and described in section 5.3.8.2 of LTR NEDC-32410P-A Supplement 1. The industry has recently trended toward the treatment of programmable hardware development as software, and the regulatory guidance that is applicable to the development of software for safety systems is also applicable to the development of programmable logic devices for safety systems. Therefore, the standard NUMAC software development process will be applied to this PLD development effort. Table 1-8 shows the mapping of the PLD development process to the life cycle phases of the standard NUMAC software development process. The process as it applies to this PLD development is summarized as follows:

Baseline 1

[[

]]

Baseline 2

[[

]]

Baseline 3

[[

]]

Baseline 4

[[

]]

Baseline 5

[[

]]

Baseline 6

[[

]]

Part 5: PRNM Platform Regulatory Compliance Discussion

NRC regulatory guidance and associated codes and standards have evolved since LTRs NEDC-32410P-A and NEDC-32410P-A Supplement 1 were first reviewed and approved by the NRC. Table 1-9 provides an evaluation of the NUMAC PRNM platform against current revisions of regulatory guidance cited in NUREG-0800 Standard Review Plan versus the guidance listed in the original LTR.

The primary NRC guideline available at the time the NUMAC design processes were developed was NRC RG 1.152 Revision 0 (1985), primarily endorsing ANSI/IEEE Std. 7-4.3.2-1982. The NUMAC design process was later evaluated against and found to be compliant with NRC RG 1.152 Revision 1 (1996), primarily endorsing IEEE Std. 7-4.3.2-1993. The latest version of NRC RG 1.152 endorses IEEE Std. 7-4.3.2-2003 and includes cyber security requirements. Table 1-10 provides a correlation of the NUMAC PRNM design process to RG 1.152 Revision 2 (2006), and specifically to the regulatory position on cyber security.

NEDC-32410P-A that was reviewed and approved by the NRC provides a comparison of the NUMAC PRNM design process to IEEE Std. 7-4.3.2-1993, which is structured as a supplement to IEEE Std. 603-1991 to identify additional requirements applicable to digital computer based safety systems. As stated in NEDC-32410P-A, IEEE Std. 603-1991 applies primarily to the overall system design and, to the extent it applies to PRNM, largely duplicates the requirements of IEEE Std. 279-1971. Clarifications in IEEE Std. 603-1991 have been considered in the evaluations of channel independence, separation, and single failures. IEEE Std. 7-4.3.2-2003 represents an incremental change from IEEE Std. 7-4.3.2-1993 that PRNM has already been evaluated against. Table 1-11 provides an updated correlation of the NUMAC PRNM design process against the requirements of IEEE Std. 7-4.3.2-2003.

Table 1-1
NUMAC Platform Changes – APRM Chassis

Module	Part Number used for Hatch APRM (1997)	Part Number used for Grand Gulf APRM (2010)
[[		
		]]

[[

]]

Table 1-2
NUMAC Platform Changes – PCI Chassis

Module	Part Number used for Hatch RBM (1997)	Part Number used for Grand Gulf PCI (2010)
[[		
		]]

[[

]]

Table 1-3
NUMAC Platform Changes – Two-Out-Of-Four Logic Module

Module	Part Number used for Hatch Two-Out-Of-Four Logic Module (1997)	Part Number used for Grand Gulf Two-Out-Of-Four Logic Module (2010)
[[		
		]]

[[

]]

Table 1-4
Changes to Hardware Modules by Parts List Revision

Module	Part Number	Parts List Rev	Date	Description
[[				

Table 1-4
Changes to Hardware Modules by Parts List Revision
(continued)

Module	Part Number	Parts List Rev	Date	Description

Table 1-4
Changes to Hardware Modules by Parts List Revision
(continued)

Module	Part Number	Parts List Rev	Date	Description

Table 1-4
Changes to Hardware Modules by Parts List Revision
(continued)

Module	Part Number	Parts List Rev	Date	Description

Table 1-4
Changes to Hardware Modules by Parts List Revision
(continued)

Module	Part Number	Parts List Rev	Date	Description

Table 1-4
Changes to Hardware Modules by Parts List Revision
(continued)

Module	Part Number	Parts List Rev	Date	Description

Table 1-4
Changes to Hardware Modules by Parts List Revision
(continued)

Module	Part Number	Parts List Rev	Date	Description

Table 1-4
Changes to Hardware Modules by Parts List Revision
(continued)

Module	Part Number	Parts List Rev	Date	Description

Table 1-4
Changes to Hardware Modules by Parts List Revision
(continued)

Module	Part Number	Parts List Rev	Date	Description

Table 1-4
Changes to Hardware Modules by Parts List Revision
(continued)

Module	Part Number	Parts List Rev	Date	Description
				]]

Table 1-5
NUMAC APRM/OPRM Firmware Changes

File	Description of Change	File Date
[[		

Table 1-5
NUMAC APRM/OPRM Firmware Changes
(continued)

File	Description of Change	File Date

Table 1-5
NUMAC APRM/OPRM Firmware Changes
(continued)

File	Description of Change	File Date

Table 1-5
NUMAC APRM/OPRM Firmware Changes
(continued)

File	Description of Change	File Date

Table 1-5
NUMAC APRM/OPRM Firmware Changes
(continued)

File	Description of Change	File Date

Table 1-5
NUMAC APRM/OPRM Firmware Changes
(continued)

File	Description of Change	File Date

Table 1-5
NUMAC APRM/OPRM Firmware Changes
(continued)

File	Description of Change	File Date

Table 1-5
NUMAC APRM/OPRM Firmware Changes
(continued)

File	Description of Change	File Date
		]]

Table 1-6.
Revision History of NUMAC Software Plans

[[		

Table 1-6.
Revision History of NUMAC Software Plans
(continued)

		]]

Table 1-7
Correlation of PRNM Design Process to BTP 7-14

Software Life Cycle Process Planning	
BTP 7-14 section B.2.1	**NUMAC PRNM Design Process**
Software Management Plan (SMP) Software Development Plan (SDP) Software Quality Assurance Plan (SQAP) Software Integration Plan (SIntP) Software Installation Plan (SInstP) Software Maintenance Plan (SMaintP) Software Training Plan (STrngP) Software Operations Plan (SOP) Software Safety Plan (SSP) Software Verification and Validation Plan (SVVP) Software Configuration Management Plan (SCMP) Software Test Plan (STP)	[[ ]]

Table 1-7
Correlation of PRNM Design Process to BTP 7-14
(continued)

Software Life Cycle Process Implementation	
BTP 7-14 section B.2.2	**NUMAC PRNM Design Process**
Requirements:	[[
• Safety analysis • V&V analysis and test reports • Configuration management reports • Testing activities	
Design:	
• Safety analysis • V&V analysis and test reports • Configuration management reports • Testing activities	

Table 1-7
Correlation of PRNM Design Process to BTP 7-14
(continued)

Implementation:	
• Safety analysis • V&V analysis and test reports • Configuration management reports • Testing activities	

Table 1-7
Correlation of PRNM Design Process to BTP 7-14
(continued)

Integration:	
• Safety analysis • V&V analysis and test reports • Configuration management reports • Testing activities	

Table 1-7
Correlation of PRNM Design Process to BTP 7-14
(continued)

Validation:	
• Safety analysis • V&V analysis and test reports • Configuration management reports • Testing activities	

Table 1-7
Correlation of PRNM Design Process to BTP 7-14
(continued)

Installation:	
• Safety analysis • V&V analysis and test reports • Configuration management reports • Testing activities	
Operations and Maintenance:	
• Safety analysis • V&V analysis and test reports • Configuration management reports • Testing activities	]]
Software Life Cycle Process Design Outputs	
BTP 7-14 section B.2.3	**NUMAC PRNM Design Process**
Software requirements specifications	[[

Table 1-7
Correlation of PRNM Design Process to BTP 7-14
(continued)

Hardware and software architecture descriptions	
Software design specifications	
Code listings	
Build documents	
Installation configuration tables	
Operations manuals	]]
Software Life Cycle Process Design Outputs	
BTP 7-14 section B.2.3	**NUMAC PRNM Design Process**
Maintenance manuals	[[
Training manuals	]]

Table 1-8
Mapping of PLD Development Process to the NUMAC Software Development Process

NUMAC Software Development Process Life Cycle Phase	PLD Development Process Life Cycle Artifacts	Method of Configuration Control
[[		
		]]

Table 1-9
Comparison of NUMAC PRNM LTR versus the Regulatory Guides Listed in the Standard Review Plan

Guide Number	Title	Revision listed in LTR	Revision listed in SRP*	Evaluation
1.22	Periodic Testing of Protection System Actuation Functions (Safety Guide 22)	Rev. 0	Rev. 0	[[
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	Rev. 0	Rev. 0	
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	Rev. 0	Rev. 2	
1.62	Manual Initiation of Protective Actions	--	Rev. 0	
1.75	Physical Independence of Electric Systems	Rev. 2	Rev. 3	
1.105	Setpoints for Safety-Related Instrumentation	Rev. 1	Rev. 3	

Table 1-9
Comparison of NUMAC PRNM LTR versus the Regulatory Guides Listed in the Standard Review Plan
(continued)

Guide Number	Title	Revision listed in LTR	Revision listed in SRP*	Evaluation
1.118	Periodic Testing of Electric Power and Protection Systems	Rev. 2	Rev. 3	
1.151	Instrument Sensing Lines	--	Rev. 0	
1.152	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants	Rev. 0	Rev. 2	

Table 1-9
Comparison of NUMAC PRNM LTR versus the Regulatory Guides Listed in the Standard Review Plan
(continued)

Guide Number	Title	Revision listed in LTR	Revision listed in SRP*	Evaluation
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	Rev. 1	
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	Rev. 0	
1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	Rev. 0	
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	Rev. 0	
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	Rev. 0	
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	--	Rev. 0	

Table 1-9
Comparison of NUMAC PRNM LTR versus the Regulatory Guides Listed in the Standard Review Plan
(continued)

Guide Number	Title	Revision listed in LTR	Revision listed in SRP*	Evaluation
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems	--	Rev. 1	
1.204	Guidelines for Lightning Protection of Nuclear Power Plants	--	Rev. 0	]]

* Applicable Regulatory Guides per Standard Review Plan Table 7-1, SAR Chapter 7.2.

Table 1-10
Correlation of PRNM Design Process to Regulatory Guide 1.152 Rev. 2

C. Regulatory Position	NUMAC PRNM Design Process
1. Functional and Design Requirements	[[
2. Security	
2.1 Concepts Phase	
2.2 Requirements Phase	
2.2.1 System Features	
2.2.2 Development Activities	
2.3 Design Phase	
2.3.1 System Features	
2.3.2 Development Activities	

Table 1-10
Correlation of PRNM Design Process to Regulatory Guide 1.152 Rev. 2
(continued)

C. Regulatory Position	NUMAC PRNM Design Process
2.4 Implementation Phase	
2.4.1 System Features	
2.4.2 Development Activities	
2.5 Test Phase	
2.5.1 System Features	
2.5.2 Development Activities	
2.6 Installation, Checkout, and Acceptance Testing	
2.6.1 System Features	
2.6.2 Development Activities	
2.7 Operation Phase	
2.8 Maintenance Phase	
2.8.1 Maintenance Activities	
2.8.2 Quality Assurance	
2.8.3 Incident Response	
2.8.4 Audits and Assessments	

Table 1-10
Correlation of PRNM Design Process to Regulatory Guide 1.152 Rev. 2
(continued)

C. Regulatory Position	NUMAC PRNM Design Process
2.9 Retirement Phase	
3. Referenced Standards	]]

Table 1-11
Correlation of PRNM Design Process to IEEE 7-4.3.2-2003

IEEE 7-4.3.2 – 2003 Paragraph	NUMAC PRNM Design Process
1. Scope	[[
2. References	
3. Definitions and abbreviations	
4. Safety system design basis	
5. Safety system criteria	
5.1 Single failure criterion	
5.2 Completion of protective action	
5.3 Quality	
5.3.1 Software development	
5.3.1.1 Software quality metrics	
5.3.2 Software tools	

Table 1-11
Correlation of PRNM Design Process to IEEE 7-4.3.2-2003
(continued)

IEEE 7-4.3.2 – 2003 Paragraph	NUMAC PRNM Design Process
5.3.3 Verification and validation	
5.3.4 Independent V&V (IV&V) requirements	
5.3.5 Software configuration management	
5.3.6 Software project risk management	
5.4 Equipment qualification	
5.4.1 Computer system testing	
5.4.2 Qualification of existing commercial computers	

Table 1-11
Correlation of PRNM Design Process to IEEE 7-4.3.2-2003
(continued)

IEEE 7-4.3.2 – 2003 Paragraph	NUMAC PRNM Design Process
5.5 System integrity	
5.5.1 Design for computer integrity	
5.5.2 Design for test and calibration	
5.5.3 Fault detection and self-diagnostics	
5.6 Independence	
5.7 Capability for test and calibration	
5.8 Information displays	
5.9 Control of access	
5.10 Repair	
5.11 Identification	
5.12 Auxiliary features	
5.13 Multi-unit stations	
5.14 Human factor considerations	
5.15 Reliability	
6. Sense and command features – functional and design requirements	
7. Execute features – functional and design requirements	
8. Power source requirements	]]

ATTACHMENT 5

GNRO-2010/00051

AFFIDAVIT SUPPORTING REQUEST TO WITHHOLD INFORMATION FROM PUBLIC DISCLOSURE

PROVIDED BY
GE-HITACHI NUCLEAR ENERGY AMERICAS, LLC
3901 CASTLE HAYNE ROAD
WILMINGTON, NC 28401

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Edward D. Schrull, state as follows:

(1) I am the Vice President, Regulatory Affairs, Services Licensing, GE-Hitachi Nuclear Energy Americas LLC (GEH). I have been delegated the function of reviewing the information described in paragraph (2) which is sought to be withheld, and have been authorized to apply for its withholding.

(2) The information sought to be withheld is contained in Enclosure 1 of GEH letter, GG-PRNM-168777-EC073, Edward Cooper (GEH) to Jon Langberg (Entergy), "Responses to Request for Additional Information - 1 and GEH Review of Proprietary Information in Grand Gulf Nuclear Station Responses to Request for Additional Information – 2 & 3," dated July 23, 2010. The GEH proprietary information in Enclosure 1, which is entitled "Responses to Request for Additional Information 1, 2, and 3," is enclosed by double square brackets. [[This sentence is an example.{3}]]. Figures containing GEH proprietary information are identified with double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit that provides the basis for the proprietary determination.

(3) In making this application for withholding of proprietary information of which it is the owner or licensee, GEH relies upon the exemption from disclosure set forth in the Freedom of Information Act (FOIA), 5 USC Sec. 552(b)(4), and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10 CFR 9.17(a)(4), and 2.390(a)(4) for trade secrets (Exemption 4). The material for which exemption from disclosure is here sought also qualifies under the narrower definition of trade secret, within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975 F2d 871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F2d 1280 (DC Cir. 1983).

(4) The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs (4)a. and (4)b. Some examples of categories of information that fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by GEH's competitors without license from GEH constitutes a competitive economic advantage over GEH and/or other companies.

b. Information that, if used by a competitor, would reduce their expenditure of resources or improve their competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information that reveals aspects of past, present, or future GEH customer-funded development plans and programs, that may include potential products of GEH.

d. Information that discloses trade secret and/or potentially patentable subject matter for which it may be desirable to obtain patent protection.

(5) To address 10 CFR 2.390(b)(4), the information sought to be withheld is being submitted to the NRC in confidence. The information is of a sort customarily held in confidence by GEH, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by GEH, not been disclosed publicly, and not been made available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to regulatory provisions or proprietary and/or confidentiality agreements that provide for maintaining the information in confidence. The initial designation of this information as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure are as set forth in the following paragraphs (6) and (7).

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, who is the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge, or who is the person most likely to be subject to the terms under which it was licensed to GEH. Access to such documents within GEH is limited to a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist, or other equivalent authority for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside GEH are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary and/or confidentiality agreements.

(8) The information identified in paragraph (2) above is classified as proprietary because it contains details developed by GEH from NEDC-32410P-A, "Nuclear Measurement Analysis and Control Power Range Neutron (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," dated October 1995. Development of the NUMAC PRNM, and information related to the design, modification, analyses methodologies and processes related to the NUMAC PRNM was achieved at a significant cost to GEH. The development of the evaluation process along with the interpretation and application of the analytical results is derived from the extensive experience database that constitutes a major GEH asset.

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to GEH's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of GEH's comprehensive BWR safety and technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology and includes development of the expertise to determine and apply the appropriate evaluation process. In addition, the technology base includes the value derived from providing analyses done with NRC-approved methods.

The research, development, engineering, analytical and NRC review costs comprise a substantial investment of time and money by GEH. The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial. GEH's competitive advantage will be lost if its competitors are able to use the results of the GEH experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to GEH would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive GEH of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing and obtaining these very valuable analytical tools.

I declare under penalty of perjury that the foregoing affidavit and the matters stated therein are true and correct to the best of my knowledge, information, and belief.

Executed on this 23rd day of July 2010.

Edward D. Schrull
Vice President, Regulatory Affairs
Services Licensing
GE-Hitachi Nuclear Energy Americas LLC
3901 Castle Hayne Rd.
Wilmington, NC 28401
edward.schrull@ge.com

ATTACHMENT 6

GNRO-2010/00051

LIST OF REGULATORY COMMITMENTS

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy will provide the human factors information requested in RAI No. 7 on or before January 17, 2011.	√		1/17/2011