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Attachment 4 contains **PROPRIETARY** information

10 CFR 50.90

GNRO-2018/00012

March 26, 2018

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

SUBJECT: Proposed Revision of Grand Gulf Nuclear Station UFSAR – Replacement of Turbine First Stage Pressure Signals with Power Range Neutron Monitoring System Signals

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

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit," Entergy Operations, Inc. (Entergy) is submitting an application for amendment to the Renewed Facility Operating License NPF-29 for Grand Gulf Nuclear Station (GGNS) Unit 1.

The proposed amendment revises the Updated Final Safety Analysis Report (UFSAR) descriptions for the replacement of the Turbine First Stage Pressure (TFSP) output signals with Power Range Neutron Monitoring System (PRNMS) output signals.

During June 2014, Entergy implemented Engineering Change (EC) 49880 in accordance with 10 CFR 50.59, "Changes, tests, and experiments," that replaced the use of the TFSP instruments with the PRNMS to measure reactor power. On December 9, 2016, the Nuclear Regulatory Commission (NRC) issued NRC Inspection Report 05000416/2016007. In this inspection report, the NRC issued non-cited violation (NCV) 050000416/2016007-02, which identified that Entergy failed to obtain a license amendment prior to implementing the proposed change. Specifically, modification EC 49880 eliminated the TFSP instrument signals to the Reactor Protection System and replaced the signals with average power range monitor signals. The NRC concluded the change reduced the diversity and resulted in a more than minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety.

Entergy has determined that the proposed change requires NRC approval per 10 CFR 50.59(c)(2). Entergy concluded that the plant modification is potentially a reduction in diversity based on the GGNS licensing basis. As such, the potential reduction in diversity is considered to be a change that results in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Attachment 1 provides an evaluation of the proposed change.

Attachment 2 provides marked-up pages of the UFSAR showing the change.

Attachment 3 provides marked-up pages of the Technical Specifications Bases showing the change.

Attachment 4 provides the proprietary GE Hitachi (GEH) Nuclear Energy Report 004N6431, Revision 1, "Technical Justification of the Grand Gulf Nuclear Station Modification to Operational Bypass Signal, Replacing Turbine First Stage Pressure with APRM Neutron Flux."

Attachment 5 provides the non-proprietary GEH Nuclear Energy Report 004N6431, Revision 1.

Attachment 6 is the affidavit signed by GEH Nuclear Energy Americas LLC, the owner of the information for Attachment 3 which contains information proprietary to GEH. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," of the Commission's regulations. Accordingly, it is respectfully requested that the information, which is proprietary to GEH be withheld from public disclosure in accordance with 2.390 of the Commission's regulations.

Approval of the proposed amendment is requested by April 8, 2019. Once approved, the amendment shall be implemented within 90 days.

In accordance with 10 CFR 50.91, "Notice for public comment; State consultation," a copy of this application, with attachments, is being provided to the designated Mississippi state official.

This letter contains no new regulatory commitments. If you have any questions or require additional information, please contact Douglas Neve at (601) 437-2103.

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 26, 2018.

Sincerely,

Eric A. Larson
Site Vice President

EAL/jh

Attachments:

1 Evaluation of Proposed Change
2 UFSAR Changes (Mark-up)
3 Technical Specifications Bases Changes (Mark-up)
4 GEH Nuclear Energy Report 004N6431, Revision 1, "Technical Justification of the Grand Gulf Nuclear Station Modification to Operational Bypass Signal, Replacing Turbine First Stage Pressure with APRM Neutron Flux" [Proprietary]
5 GEH Nuclear Energy Report 004N6431, Revision 1, "Technical Justification of the Grand Gulf Nuclear Station Modification to Operational Bypass Signal, Replacing Turbine First Stage Pressure with APRM Neutron Flux" [Non-Proprietary]
6 Affidavit

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cc: without Attachments

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Attachment 1 to GNRO-2018/00012

Evaluation of the Proposed Change

Evaluation of the Proposed Change

Subject: Proposed Revision of Grand Gulf Nuclear Station UFSAR – Replacement of Turbine First Stage Pressure Signals with Power Range Neutron Monitoring System Signals

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1.0 SUMMARY DESCRIPTION

The proposed amendment to Renewed Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1 (GGNS) revises the Updated Final Safety Analysis Report (UFSAR) to reflect the permanent replacement of Turbine First Stage Pressure (TFSP) output signals with the Power Range Neutron Monitoring System (PRNMS) output signals that control safety related functions including Low Power Setpoint (LPSP) and High Power Setpoint (HPSP), Turbine Stop Valve (TSV) closure and Turbine Control Valve (TCV) fast closure scram enable/bypass, and End of Cycle Recirculation Pump Trip (EOC-RPT) enable/bypass. Additionally, the non-safety related functions for Feedwater Low Power Set-Down, Hydrogen Water Chemistry Trips, and Main and Reheat Steam Systems make use of PRNMS signals rather than TFSP signals. The non-safety related functions are discussed here for information only and are not considered as a part of the proposed amendment.

This proposed amendment eliminates the potential for a transient caused by the mechanical failure of the TFSP sensing lines and instruments. It also eliminates process delays in the steam lines as the PRNMS voltage output signals are based on Average Power Range Monitoring (APRM) signals, a direct and immediate measurement of neutron flux. The PRNMS signals are divisionally separated, safety-related and provide reliability, quality and defense-in-depth that the TFSP sensing lines and instruments could not provide. The replacement of the TFSP output signals with the PRNMS output signals enhances plant safety and improves reliability.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

Neutron Monitoring System/Average Power Range Monitoring Subsystem/ Power Range Neutron Monitoring System

In License Amendment No. 188 (Reference 6.1), the Nuclear Regulatory Commission (NRC) issued revisions to the GGNS Technical Specification (TSs) to reflect replacement of the existing APRM subsystem of the Neutron Monitoring System with a digital General Electric Hitachi Nuclear Measurement Analysis and Control (NUMAC) PRNMS. The PRNMS was incorporated into the GGNS license basis under engineering change (EC) 21999, "GGNS EPU Power Range Neutron Monitoring System NUMAC Upgrade," and was installed during Refueling Outage 18 (Spring 2012).

The Neutron Monitoring System is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. The source range monitors (SRMs) and the intermediate range monitors (IRMs) provide flux level indications during reactor startup and low power operation. The local power range monitors (LPRMs) and APRMs allow assessment of local and overall flux conditions during power range operation. The traversing in-core probe system (TIP) provides a means to calibrate the individual LPRM sensors. The Neutron Monitoring System provides inputs to the Rod Control and

Information System (RCIS) to initiate rod block trips if preset flux limits are exceeded, and inputs to the Reactor Protection System (RPS) to initiate a scram if other limits are exceeded.

The PRNM channels receive input signals from the LPRM channels and provide a continuous indication of average reactor power from a few percent to greater than rated reactor power. The APRM subsystem has sufficient redundant channels to meet industry and regulatory safety criteria. Under the worst permitted input LPRM bypass conditions, the APRM subsystem is capable of generating a trip scram signal before the average neutron flux increases to the point that fuel damage is probable. The Oscillation Power Range Monitor (OPRM), a sub-function of the APRM, is also capable of generating a trip scram signal to terminate reactor operation in states that could result in development of power oscillations due to neutronic/ thermal-hydraulic instability. The outputs from all four PRNM channels go to four independent 2-out-of-4 logic voter modules. Each of the 2-out-of-4 logic modules interfaces with one of the four RPS input channels (A1, A2, B1, and B2). The trip outputs from all four PRNM channels are sent to each 2-out-of-4 logic module, such that each input sent to RPS is a voted result of all four PRNM channels. A trip output to RPS is provided when at least two of the same type of trip inputs is in a tripped state (eg. 2 PRNM channels have an APRM or OPRM upscale/INOP trip). Any one APRM can initiate a rod block, regardless of the Reactor Mode switch position. The APRM upscale rod block and the thermal power scram trip set points vary as a function of reactor recirculation driving loop flow. The APRM flux signal is passed through a filter for the thermal power scram trip. The filter is a 6 second time constant to simulate thermal power. A faster response time APRM upscale trip has a fixed set point, not variable with recirculation flow. Any APRM upscale or inoperative trip from any one unbypassed APRM channel will result in a "half-trip" in all four 2-out-of-4 logic modules, but no trip outputs to either RPS trip system. A trip of the APRM Neutron Flux - High, Setdown; Fixed Neutron - High; INOP; or Flow Biased Stimulated Thermal Power - High function from any two unbypassed APRM channels will result in a full trip in each 2-out-of-4 logic module, which in turn results in two trip inputs into each RPS trip system logic channel (A1, A2, B1, and B2). The system allows the operator to bypass the trips from one PRNM channel, but no voter channels can be bypassed.

Rod Control and Information System (RCIS)

Control rod patterns and associated control rod reactivity worths are regulated by the RCIS. This system utilizes redundant inputs to provide rod pattern control over the complete range of reactor operations. The control rod worths are limited to such an extent that the rod drop accident and the power range rod withdrawal error become unimportant. The RCIS provides for stable control of core reactivity in both the single rod or rod gang mode of operation. The Bank Position mode of RCIS provides protection from a rod drop accident from startup to the low power setpoint (LPSP). The Rod Withdrawal Limiter provides protection from the rod withdrawal error for all conditions above the LPSP.

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the Rod Withdrawal Limiter provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod pattern controller (RPC) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident.

The purpose of the Rod Withdrawal Limiter is to limit control rod withdrawal to preclude a minimum critical power ratio Safety Limit violation. The Rod Withdrawal Limiter supplies a trip signal to the RCIS to appropriately inhibit control rod withdrawal during power operation equal to or greater than the LPSP. The Rod Withdrawal Limiter has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. The rod block logic circuitry in the RCIS is arranged as two redundant and separate logic circuits. These circuits are energized when control rod movement is allowed. The output of each logic circuit is coupled to a comparator by the use of isolation devices in the rod drive control cabinet. The two logic circuit signals are compared and rod blocks are applied when either circuit trip signal is present. Control rod withdrawal is permitted only when the two signals agree. Each rod block logic circuit receives control rod position indication from a separate channel of the Rod Position Information System, each with a set of reed switches for control rod position indication. Control rod position is the primary data input for the Rod Withdrawal Limiter. The PRNMS is used to determine reactor power level, with an LPSP and a high power setpoint (HPSP) used to determine allowable control rod withdrawal distances. Below the LPSP, the Rod Withdrawal Limiter is automatically bypassed.

The purpose of the RPC is to ensure control rod patterns during startup are such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% rated thermal power (RTP). The sequences effectively limit the potential amount and rate of reactivity increase during a control rod drop accident. The RPC, in conjunction with the RCIS, will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the specified sequence. The rod block logic circuitry is the same as that described above. The RPC uses the power range neutron monitoring system to determine when reactor power is above the power at which the RPC is automatically bypassed.

LPSP, HPSP and Low Power Alarm Point (LPAP) Setpoint Functions

The RCIS receives permissive trip signals at reactor power levels corresponding to LPSP and HPSP that activate the Banked Position Withdrawal Sequence and Rod Withdrawal Limiter functions of the RPC to provide the proper constraints for movement of control rods. The LPAP provides an alarm during power decrease that the LPSP is approaching and the RCIS logic will change from Rod Withdrawal Limiter to Banked Position Withdrawal Sequence. The LPSP and HPSP permissive trip signals are referred to as "Power Setpoints" and LPAP is referred to as the "Alarm" setpoint in RCIS documentation.

<u>Turbine Generator to Reactor Protection System Interface</u>

Two conditions which initiate reactor scram are TSV closure and TCV fast closure when reactor power is above a preselected percent of rated power. The APRM neutron flux output is provided for bypassing the stop valve closure and control valve fast closure inputs at low power levels.

In the original plant design, the TCV fast closure scram and TSV closure scram are automatically bypassed if reactor power is low (below the bypass setpoint). Closure of these valves below a low initial power level does not threaten the integrity of any radioactive material release barrier. Turbine control valve fast closure and TSV closure trip bypass is effected by four transmitters associated with the APRM neutron flux. Any one channel in a bypass state produces a control room annunciation. No single failure of a transmitter can prevent a TSV closure scram or TCV fast closure scram. In addition, this bypass is automatically removed when the reactor power exceeds the bypass reset point. This reset setpoint is established to ensure the bypass is removed prior to reactor power exceeding the Analytical Limit of 35.4% RTP.

The turbine stop valve closure trip and control valve fast closure trip operating bypass complies with the single-failure criterion. Wiring from the PRMN signal converter is routed in conduit to the RPS cabinets in the control room.

TSV Closure Scram, TCV Fast Closure Scram and EOC-RPT Bypass Setpoint Functions

Bypass signals exist for the TCV fast closure and TSV closure functions and, for EOC-RPT subsystem in the event of TCV fast closure and TSV closure, when the reactor power is less than a prescribed reactor power level. The bypass signal prevents unnecessary reactor scrams (and EOC-RPT) when the reactor power is low enough to not challenge the Safety Limit Minimum Critical Power Ratio (MCPR) when these turbine control and stop valve closure events occur (due to main turbine trip or generator load rejection).

2.2 Reason for the Proposed Change

The TFSP sensing lines have failed multiple times during prior operating cycles. Repairs including the installation of temporary modifications were performed to address the failures. Provided below is a summary of the history of the failures.

During power operation (1995), an isolation valve to the TFSP transmitter developed a leak.

During Refueling Outage 8 startup (1996), a weld attaching the TFSP sensing line to the main steam line developed a leak.

During Refueling Outage 9 startup (1998), a weld repair of the TFSP sensing line separates from the main steam line. Additionally, during startup several compression fitting repairs were conducted.

During Refueling Outage 10 startup (12/12/99), the TFSP sensing line separated at the tube fitting on the main steam line side of the pressure transmitter isolation valve.

During Refueling Outage 14 startup (10/19/2005), the TFSP sensing line for division 2 failed. The tubing sheared at the toe of the weld to the tube adapter.

In Refueling Outage 16 (10/27/08), flexible hoses were installed in division 1 and 2, main steam line "A" and main steam line "B", TFSP sensing lines.

During Cycle 17 (3/20/10), approximately one month before Refueling Outage 17, the division 1 main steam line "B" flexible hose failed.

During Refueling Outage 17 (5/20/10), both flexible hoses were removed and tubing was installed utilizing the best previous design that had lasted with no failure for over two years before they were removed to install the flexible connections. Additionally, during Refueling Outage 17, the TFSP sensing lines and main steam lines were instrumented with accelerometers to measure vibration inputs and response during startup periods when vibration is known to be most severe.

During Cycle 18 (6/7/10), approximately two months after Refueling Outage 17 startup, the TFSP sensing line for division 2 failed. Main steam line "A" tubing sheared at the toe of the weld to the tube adapter. Corrective actions included a temporary modification to install bypass signals for "B" TFSP transmitters and a temporary leak repair to control the leak until Refueling Outage 18.

In Refueling Outage 18 (4/15/11), an engineering change maintained the first stage pressure transmitters tap location on the main steam inlet piping, but relocated the branch connections to an area of lower vibration levels to reduce the fatigue stresses encountered with the current location and installed three-quarter inch piping up to the first anchor.

During Refueling Outage 19 startup on March 17, 2014, with the plant at 41% RTP, a manual reactor scram was initiated due to a steam leak in the turbine building. The steam leak in the turbine building was the result of a failed TFSP sensing line followed by a failed main steam line four inch drain line. The failed drain line was replaced and both TFSP sensing lines were removed. Temporary modifications implemented "a current source" strategy that removed the TFSP sensing lines and installed two electronic modules as an interim action while a permanent design change to eliminate the need for the TFSP sensing lines was being developed.

In June 2014, Entergy implemented modification EC 49880 in accordance with 10 CFR 50.59 that replaced the use of the TFSP sensing line and instruments with the PRNMS to measure reactor power.

On December 9, 2016, the NRC issued Baseline Inspection Report 05000416/2016007 (Reference 6.2), dated December 9, 2016, that included Severity Level IV, Non-Cited Violation (NCV) 05000416/2016007-02, "Failure to obtain NRC approval for changes to the Reactor Protection System." The NCV states, in part

> Enforcement. Title 10 CFR 50.59(c)(2) states, in part, that "a licensee may make changes in the facility as described in the final safety analysis report, make changes in the procedures as described in the final safety analysis report, and conduct tests or experiments not described in the final safety analysis report without obtaining a license amendment only if...the change, test, or experiment does not meet any of the criteria in paragraph (c)(2) of this section." Paragraph (c)(2), states, in part, "a licensee shall obtain a license amendment prior to implementing a proposed change, test, or experiment if the change, test, or experiment would ... result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component, important to safety previously evaluated in the final safety analysis report." Contrary to this requirement, from June 24, 2014, until November 3, 2016, the licensee made a change to the facility as described in the final safety analysis report that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety. Specifically, modification Engineering Change 49880 eliminated the turbine first stage pressure instruments signals to the reactor protection system and replaced the signals with

average power range monitor signals which reduced the diversity, separation, and independence, and resulted in a more than increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety.

Therefore, Entergy is submitting a license amendment request and requesting NRC approval of the permanent plant modification per 10 CFR 50.59(c)(2).

2.3 Description of the Proposed Change

In previous operating cycles, the TFSP sensing lines have failed multiple times requiring temporary modifications to be installed to maintain a valid input signal to maintain functions of the RPS and the RCIS. The TFSP is used by these systems as an indication of Reactor Core Thermal Power to control various functions including LPSP and HPSP, TSV closure and TCV fast closure scram enable/bypass, and EOC-RPT enable/bypass.

During Refueling Outage 18, the new PRNMS was implemented which has an available safety related analog output based on APRM power levels that provide a more accurate indication of Reactor Core Thermal Power. As the electronic signals are not susceptible to mechanical failure of the sensing line, a permanent plant modification was developed to take advantage of the available signals from the PRNMS and use these as inputs to perform those functions previously performed by the TFSP sensing lines and instruments. Section 3.0 describes the permanent plant modification.

The following UFSAR Sections, Table, and Figures are changed to reflect the permanent plant modification. Attachment 2 provides the markups of the UFSAR pages.

Section 7.1.2.7, Safety System Settings
Section 7.2.1.1.4.4.2, Turbine Stop Valve and Turbine Control Valve Fast Closure
Section 7.2.2.1.2.3.1.2, Single Failure Criterion (IEEE Std. 279-1971, paragraph 4.2)
Section 7.2.2.1.2.3.1.8, Derivation of System Inputs (IEEE Std. 279-1971, paragraph 4.8)
Section 7.2.2.1.2.3.1.10, Capability for Test and Calibration (IEEE Std. 279-1971, paragraph 4.10)
Table 7.2-1 (sheet 2 of 2), Reactor Protection System Instrumentation Specifications
Figure 7.2-001B
Figure 7.2-001C
Section 7.6.1.7.3, Equipment Design
Section 7.6.1.8.3.2, Logic
Section 7.7.1.5.3.4.3, Turbine Generator to Reactor Protection System Interface
Section 15.2.3.2.2.3, Turbine Trip at Low Power w/o Bypass
Section 15.2.3.3.3.3, Turbine Trip w/o Bypass, Low Power

3.0 TECHNICAL EVALUATION

This evaluation establishes the justification and bases for using neutron flux signals from the four APRMs in PRNMS to replace TFSP signals from pressure transmitters currently used as inputs representing reactor power for the safety related functions including LPSP and HPSP, TSV closure and TCV fast closure scram enable/bypass, and EOC-RPT enable/bypass. Further justification is included in GE Hitachi (GEH) Nuclear Energy Report 004N6431, Revision 1, "Technical Justification of the Grand Gulf Nuclear Station Modification to Operational Bypass Signal, Replacing Turbine First Stage Pressure with APRM Neutron Flux," (Reference 6.3).

This report is provided in Attachment 3. The replacement of the TFSP output signals with the PRNMS output signals enhances plant safety and improves reliability.

Description of Permanent Plant Modification

The original GGNS design used pressure transmitters to sense the TFSP, and each transmitter sent a 4 – 20 mA signal proportional to the pressure to master/slave trip units. The trip units then sent the LPSP, HPSP and LPAP permissive trip signals to the RCIS and bypass interlock signals for RPS scram and EOC-RPT. The TFSP pressure versus RTP calibration was based on the turbine heat balance. The TFSP to reactor power correlation is subject to error introduced by various factors such as turbine bypass valve leakage, steam flow in the turbine reheaters, reduced feedwater heating, changes in condenser vacuum, etc. Thus, this indirect inference of reactor power from the TFSP measurements is inherently less accurate than the direct measurement of reactor power from the APRM measurement. In summary with the TFSP configuration, once the setpoints are fixed in terms of pressure, any changes in the reactor power to TFSP correlation will affect the reactor power corresponding to the setpoints, and that results in setpoint errors which are accounted for operationally.

The permanent plant modification replaced the TFSP measurement with a direct neutron flux power measurement from the APRMs. Each APRM sends a 0 – 10 volt signal proportional to 0-125% reactor power to a voltage-to-current converter that generates a 4 to 20 mA current like the TFSP pressure transmitters. These signals then go to the same Master/Slave Trip unit and LPSP and HPSP permissive trip signals, and RPS (and EOC-RPT) bypass signals would be generated and sent to the RCIS, RPS and RPT systems in the same way. The APRMs are calibrated against reactor thermal heat balance in accordance with operations procedure 03-1-01-1 and 06-RE-1C51-W-0001 when there is a valid core heat balance and provide an accurate measurement of reactor power. Note that the APRMs are already used to provide RPS scram signals for various TS setpoints spanning a wide power range from approximately 10% RTP (APRM setdown setpoint) to approximately 120% RTP (APRM high power setpoint).

The physical plant changes were implemented by installing signal converters to convert the safety related 0 to 10 VDC outputs from PRNMS to safety related 4 to 20 mA DC signals useable by the existing safety related trip units.

The APRM's output safety related 0-10VDC signals that represent 0-125% RTP. The existing trip units require a 4 to 20 mA signal input. The modification installed Moore Industries voltage-to-current converters in the instrument loops. These converters were located in the associated termination cabinets in the Control Room and Upper Cable Spreading Room. Installation of the converters into the instrument loops is justified based on the revisions to the uncertainty and scaling calculations. The modification installs new instrument cables between the existing Control Room PRNMS cabinets and the termination cabinets. This is done in accordance with the applicable panel and cable termination specifications.

The existing PRNMS has safety related as well as isolated non-safety related power signals available for use. The circuits conform to all required separation criteria. The modification has been designed in accordance with the applicable channel redundancy, independence and separation criteria. Therefore, there are no adverse interactions between safety related and non-safety related systems or components.

The APRM signals generated within each PRNMS channel can fail low, fail high, or fail as-is. These are the same failure modes as the signals from the existing pressure transmitters. While the mechanisms which can cause APRM signal failures are different than those which can cause TFSP signal failures, the results are the same. With respect to the RPS and RCIS functions affected by the proposed modification, the worst case is for the signals to fail low with the reactor at high power resulting in inappropriate RCIS interlocks and bypassing of RPS trips. The results of this worst-case scenario are unchanged by the modification.

There is a conservative operational difference when operating with APRM neutron flux rather than TFSP signal, if the TSV/TCV trip occurs at a power lower than the RPS bypass setpoint and the Turbine Bypass Valves remain closed. For these scenarios, the APRM neutron flux configuration could go out of bypass quickly because the APRM neutron flux value would quickly increase above the setpoint, and the reactor would scram from the TSV/TCV closure trip signals as soon as the neutron flux reached the scram bypass setpoint. However, the TFSP configuration would remain in bypass because the measured TFSP would stay below the scram bypass setpoint, and the reactor would only scram when the high neutron flux (or high reactor dome pressure) setpoint is reached. Thus, for these scenarios, the chance of spurious trip could be higher for the APRM neutron flux configuration than the TFSP configuration, although the earlier scram by the APRM neutron flux configuration is in the safe direction.

The core operating limits report as defined in Technical Specification 1.1 was not affected as a result of replacing TFSP signals with PRNMS signals for measuring reactor power.

<u>Diversity (APRMs vs. TFSP)</u>

Entergy Letter GNRO-2010/0056 (Reference 6.4) identified that the instrument setpoint methodology currently implemented at GGNS is based on Instrument Society of America (ISA) Standard 67.04 Part II, 1994, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation" (Reference 6.5), and the General Electric Hitachi (GEH) Instrument Setpoint Methodology (ISM) specified in NEDC-31336P-A, "General Electric Instrument Setpoint Methodology" (Reference 6.6). Additionally, in the NRC safety evaluation for Amendment No. 191 (Reference 6.7), the NRC states: "Entergy stated that its setpoint calculations in support of the Extended Power Uprate (EPU) license amendment request (LAR) were based on NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," September 1996 (Reference 120), which includes the NRC-approved safety evaluation dated November 6, 1995.

In the NRC Cover letter and safety evaluation associated with NEDC-31336, the NRC states, in part:

> "GE Topical Report NEDC-31336 provides an important reference for understanding the GE methodology for selection of instrumentation setpoints. The topical report demonstrates the methodology used by GE and satisfactorily addresses setpoint issues previously identified by the staff.
>
> However, because the topical report is limited to the development of a few sample calculations, it is not to be used by any plant as the sole basis for individual, plant specific setpoints. That is, each plant must provide its own plant unique analysis for the setpoints. The examples given in the topical report are used by GE only to show the safety margins and typical channel errors that might be expected. Since plants have different instruments, environments, seismic and other requirements, only examples have been provided by GE."

Plant specific calculations are utilized for determining the instrumentation setpoints based on the methodology in NEDC-31336 P-A.

NEDC-31336 P-A, Section 3.25, states, in part: "Turbine first stage pressure has been historically used as the parameter to approximate reactor power and effect the actual trip bypass. The RPS design purposely chooses this parameter, as opposed to the more direct measurement of power such as neutron flux, in order to assure diversity between the TSVC [turbine stop valve closure] and TCVFC [turbine control valve fast closure] scram functions and the neutron flux scram function."

The UFSAR (Section 7.2.1.1.4.2, paragraph d) states the following about TSV closure.

> "Diversity of trip initiation for increases in reactor vessel pressure due to termination of steam flow by turbine stop valve or control valve closure is provided by reactor vessel high-pressure trip signals. A closure of the turbine stop valves or control valves at steady-state conditions would result in an increase in reactor vessel pressure. If a scram was not initiated from these closures, a scram would occur from high reactor vessel pressure. Reactor vessel high pressure is an independent variable and for this condition provides diverse trip initiating circuits for the protective action (scram)."

In the discussion about the TCV fast closure, the UFSAR (Section 7.2.1.1.4.2 paragraph e) simply points back to the discussion just quoted.

> "The discussion of diversity for turbine control valve fast closure is the same as that for turbine stop valve closure provided in subsection 7.2.1.1.4.5 and paragraph d. above."

Also, when the UFSAR discusses redundancy and diversity of RPS scram inputs (Section 7.2.1.1.4.5), it identifies TSV closure and TCV fast closure scrams as anticipatory of reactor vessel high-pressure scram, but not anticipatory of neutron flux scram.

> "[The] main steam line isolation valve closure, turbine stop valve closure, and turbine control valve fast closure are anticipatory of a reactor vessel high-pressure and are separate inputs to the system."

Although NEDC-31336 P-A discusses the turbine first stage pressure as a method for achieving diversity, the NRC safety evaluation states that all plants must perform their own analysis. The GGNS UFSAR describes the drywell high pressure as providing the required diversity.

Additional information is available to plant operators to determine whether the scram bypass has been lifted at the correct power. The information available to plant operators includes APRM required calibration checks and the scram bypass power level and low-pressure alarm are annunciated in the control room. Entergy is submitting this license amendment request and requesting NRC approval of the permanent plant modification per 10 CFR 50.59(c)(2). It was concluded that the potential reduction in diversity is considered to be a change that results in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the UFSAR.

Branch Technical Position 7-19 Review

The NUMAC PRNMS topical report (Reference 6.8) addressed diversity by generically identifying diverse trip functions for UFSAR events and requiring licensees to make a statement of applicability. During NRC review of the GGNS LAR for implementing the NUMAC PRNMS (Reference 6.9), the NRC staff issued a series of Requests for Additional Information (RAIs) concerning diversity and defense-in-depth. The scope of some of the RAIs involved demonstrating that the modification to install the NUMAC PRNMS met the acceptance criteria in NRC BTP 7-19, "Guidance for Evaluation of Diversity and Defense in Depth in Digital Computer-based Instrumentation and Control Systems," (Reference 6.10). The conclusion in the RAI response (Reference 6.11) is that GGNS, with NUMAC PRNMS installed, meets the acceptance criteria. Additionally, in the NRC safety evaluation (Reference 6.1), the NRC staff determined that the proposed change provides sufficient diversity and defense-in-depth to satisfy the acceptance criteria.

Subsequently, GGNS submitted an LAR for Maximum Extended Load Line Limit Analysis Plus (MELLLA+) (Reference 6.12). As part of that review process, the NRC staff issued Requests for Supplemental Information (RSIs), including an RSI about diversity and defense-in-depth. In the GGNS response (Reference 6.13), content from the previously submitted RAI response (Reference 6.11) was revisited in light of MELLLA+ operations. It was found that MELLLA+ did not alter the conclusions about diversity and defense-in-depth, and the MELLLA+ LAR was subsequently approved in Amendment No 205 (Reference 6.14).

GEH Nuclear Energy Report 004N6431, Revision 1 (Reference 6.3), reviews each of the nine criteria in BTP 7-19 as it relates to the modification to replace the TFSP instrumentation with the APRMS. The GEH report is included as Attachments 3 and 4 in this LAR. This review in section 4 of the GEH report concludes that either the criterion was satisfied or not applicable to this modification.

Instrumentation Changes and Uncertainties

RCIS LPSP

GGNS Setpoint calculation JC-Q1C11N654-2, "Instrument Loop Uncertainty and Setpoint for Rod Pattern Controller Low Power Setpoint (Banked Rod Withdrawal and Rod Worth Limiter Functions)," has been revised to demonstrate that the system as modified with the new PRNMS signals can support the existing Allowable Values in accordance with the existing setpoint methodologies. This calculation determines the RPC LPSP and Low Power Reset for the 1C11N654A and 1C11N654B instrument loops and the lower and upper Allowable Values that they protect. The RCIS LPSP lower Allowable Value of ≥ 10% and upper Allowable Value of ≤ 35% are listed in the GGNS TS SR 3.3.2.1.5. This modification does not change these Allowable Values.

The existing RCIS LPSP upper and lower Analytical Limits originally established by GEH, 36% RTP and 8% RTP respectively, are not listed in the TSs or the Technical Requirements Manual. These Analytical Limits have been adjusted to 38% RTP and 5% RTP respectively so that the modified system may be calibrated within the Allowable Values given the PRNMS and signal converter uncertainties. Justification for the Analytical Limit changes have been provided by GEH and is summarized as follows:

- The existing lower Analytical Limit was chosen to be 8%; however, there is no analytical basis for this number. The existing GEH design basis prescribes a 5% RTP value. This plant modification changes the lower Analytical Limit to match the GEH design basis.

- There is enough conservatism in the unblocked rod withdrawal analysis that changing the upper Analytical Limit from 36% to 38% will not challenge the safety limit MCPR.

The assumptions used in calculation JC-Q1C11N654-2 are as follows:

1. All uncertainties are two standard deviations (2σ) unless otherwise specified.

2 M&TE accuracy is assumed to be equal to the larger of the device reference accuracy or the setting tolerance.

3. The APRMs, calibrated weekly, measure average reactor power and provide an unfiltered output referred to as Neutron Flux. The Neutron Flux output is prompt and follows changes in thermal power promptly. For the purposes of this calculation, Neutron Flux will be considered to be equivalent to the RTP.

4. Insulation Resistance Effects are negligible because the design basis event for the loop function does not result in harsh environmental conditions, and the normal environment is relatively mild. Therefore IR = ± 0.0.

5. In these calculations, the Rosemount 510DU Trip Unit "Normal/Normal" Repeatability specification is used as the Reference Accuracy (RA_2), and the "Adverse/Normal" Repeatability specification is used as the trip unit uncertainty (A_2). This is conservative because both of these values from Rosemount include Drift for 6 months in addition to the Reference Accuracy and environmental uncertainties.

   The Normal/Normal Repeatability is ±0.13% Span. The "Adverse/Normal" and "Normal/High" Repeatability is ±0.20% Span.

   The "Normal/Normal" Repeatability (±0.13% Span) best represents RA_2 because calibrations are performed under stable environmental conditions. The "Adverse/Normal" Repeatability (±0.20% Span) includes expanded environmental limits and is conservatively used for the A_2 uncertainty parameter.

   The accuracy of the trip units is ±0.2% span = ±0.2% * 125 % RTP = ±0.25% RTP including 6 months of drift and adverse environmental circumstances. However, the Normal/Normal repeatability specification ±0.13% span = ±0.13% * 125 % RTP = ±0.1625% RTP better reflects the conditions during calibrations and the ALT is calculated using it.

   Note that the even though the Normal/Normal repeatability specification allows for 6 months of drift, it is assumed to be the reference accuracy.

   For purposes of calculating the setpoint and reset, MTE = RA is used as it is more conservative than the actual MTE.

6. The trip units are calibrated every 92 days (3 months). The trip unit repeatability specifications include 6 months of drift. The 6 month specification is conservatively used for all calibration periods less than or equal to 6 months.

7. The Rosemount Model 510DU model is obsolete, and these units may be replaced by the Model 710DU in the future. The performance specifications for the 710DU are equal to or better than those of the 510DU.

8. No Drift is specified by the signal converter vendor. The vendor published Reference Accuracy will be assumed to be applicable as Drift over a period of 6 months.

9. Radiation effects are negligible and set to zero. Since all of the instruments are located in the Control Room, the design basis for the instrument loops does not result in harsh environmental conditions, and the normal environment is mild, including no exposure to radiation.

The below tables provide the results of calculation JC-Q1C11N654-2.

PARAMETER	INSTRUMENTS A/B % RTP	PARAMETER	INSTRUMENTS A/B % RTP
Upper Analytical Limit (AL_u)	38		
Calculated Upper Allowable Value (AV_{uCALC})	35.065	Actual Upper Allowable Value (AV_u)	35.0
Calculated Nominal Trip Reset ($NTRP_{CALC}$)	35.003	Maximum Nominal Trip Reset ($NTRP_{MAX}$)	≤30.00
		Nominal Trip Reset (NTRP)	26.0
Calculated Nominal Trip Setpoint ($NTSP_{CALC}$)	7.997	Minimum Nominal Trip Setpoint ($NTSP_{MIN}$)	≥15.0
		Nominal Trip Setpoint (NTSP)	25.0
Calculated Lower Allowable Value (AV_{lCALC})	7.935	Actual Lower Allowable Value (AV_l)	10
Upper Analytical Limit (AL_l)	5		

PARAMETER	VALUE
Instrument Loops (Circuits)	1C71N654A & B
Loop Uncertainty (LU)	±2.935% RTP
Total Loop Uncertainty (TLU)	±2.997% RTP
Loop Drift (D_L)	±0.6085% RTP
Loop MTE Uncertainty (C_L)	±1.697% RTP
APRM ALT (ALT_1)	±0.132 Volts
Converter ALT (ALT_C)	±0.016 mA
Trip Unit ALT (ALT_2)	±0.021 mA
APRM AFT (AFT_1)	±0.04 Volts
Converter AFT (AFT_C)	±0.036 mA
Trip Unit AFT (AFT_2)	±0.03 mA
As-Left Loop Tolerance (ALT_L)	±1.658 % RTP (±0.21 mA)
As-Found Loop Tolerance (AFT_L)	±0.6081 % RTP (±0.077 mA)

The NTRP and the NTSP are conservative with respect to the upper and lower Analytical limits (respectively).

RCIS HPSP

Calculation JC-Q1C11N654-3, "Instrument Loop Uncertainty and Setpoint for Rod Pattern Controller, Rod Worth Limiter Function, High Power Setpoint (HPSP)," determines the RCIS HPSP and reset for the 1C11N654C and 1C11N654D instrument loops. The RCIS HPSP Analytical Limit (> 70% RTP) is listed in TS SR 3.3.2.1.6 and is not being changed by this plant modification. The RCIS HPSP Allowable Value is not listed in the TSs or in the Technical Requirements Manual. The Allowable Value is determined by calculation JC-Q1C11N654-3, which has been revised by this modification to account for the uncertainties associated with PRNMS and the new signal converters. This calculation revision establishes a new Allowable Valve of 66% RTP and demonstrates that the nominal trip setpoint of 62% RTP is conservative. The NTSP value was increased from its current value to provide improved operating margin, and still have sufficient margin to the Allowable Value and Analytical Limit to meet to GEH setpoint methodology requirements.

The assumptions used in calculation JC-Q1C11N654-3 are as follows:

1. All uncertainties are two standard deviations (2σ) unless otherwise specified.

2. M&TE accuracy is assumed to be equal to the larger of the device reference accuracy or the setting tolerance.

3. The APRMs, calibrated weekly, measure average reactor power and provide an unfiltered output referred to as Neutron Flux. The Neutron Flux output is prompt and follows changes in thermal power promptly. For the purposes of this calculation, Neutron Flux will be considered to be equivalent to the RTP.

4. Insulation Resistance Effects are negligible because the design basis event for the loop function does not result in harsh environmental conditions, and the normal environment is relatively mild. Therefore IR = ± 0.0.

5. In these calculations, the Rosemount 510DU Trip Unit "Normal/Normal" Repeatability specification is used as the Reference Accuracy (RA_2), and the "Adverse/Normal" Repeatability specification is used as the trip unit uncertainty (A_2). This is conservative because both of these values from Rosemount include Drift for 6 months in addition to the Reference Accuracy and environmental uncertainties.

   The Normal/Normal Repeatability is ±0.13% Span. The "Adverse/Normal" and "Normal/High" Repeatability is ±0.20% Span.

   The "Normal/Normal" Repeatability (±0.13% Span) best represents RA_2 because calibrations are performed under stable environmental conditions. The "Adverse/Normal" Repeatability (±0.20% Span) includes expanded environmental limits and is conservatively used for the A_2 uncertainty parameter.

   The accuracy of the trip units is ±0.2% span = ±0.2% * 125 % RTP = ±0.25% RTP including 6 months of drift and adverse environmental circumstances. However, the Normal/Normal repeatability specification ±0.13% span = ±0.13% * 125 % RTP = ±0.1625% RTP better reflects the conditions during calibrations and the ALT is calculated using it.

Note that the even though the Normal/Normal repeatability specification allows for 6 months of drift, it is assumed to be the reference accuracy.

For purposes of calculating the setpoint and reset, MTE = RA is used as it is more conservative than the actual MTE.

6. The trip units are calibrated every 92 days (3 months). The trip unit repeatability specifications include 6 months of drift. The 6 month specification is conservatively used for all calibration periods less than or equal to 6 months.

7. The Rosemount Model 510DU model is obsolete, and these units may be replaced by the Model 710DU in the future. The performance specifications for the 710DU are equal to or better than those of the 510DU.

8. No Drift is specified by the signal converter vendor. The vendor published Reference Accuracy will be assumed to be applicable as Drift over a period of 6 months.

9. Radiation effects (RE) are negligible and set to zero. Since all of the instruments are located in the Control Room, the design basis for the instrument loops does not result in harsh environmental conditions, and the normal environment is mild, including no exposure to radiation

The below tables provide the results of calculation JC-Q1C11N654-3.

PARAMETER	INSTRUMENTS C/D % RTP	PARAMETER	INSTRUMENTS C/D % RTP
Analytical Limit (AL)	70		
Calculated Allowable Value (AV_{CALC})	67.065	Actual Allowable Value (AV)	66.0
Calculated Nominal Trip Setpoint ($NTSP_{CALC}$)	67.03	Actual Nominal Trip Setpoint (NTSP)	62.00

PARAMETER	VALUE
Instrument Loops (Circuits)	1C71N654C & D
Loop Uncertainty (LU)	±2.935% RTP
Total Loop Uncertainty (TLU)	±2.997% RTP
Loop Drift (D_L)	±0.6085% RTP
Loop MTE Uncertainty (C_L)	±1.697% RTP
APRM ALT (ALT_1)	±0.132 Volts
Converter ALT (ALT_C)	±0.016 mA
Trip Unit ALT (ALT_2)	±0.021 mA
APRM AFT (AFT_1)	±0.04 Volts
Converter AFT (AFT_C)	±0.036 mA
Trip Unit AFT (AFT_2)	±0.03 mA
As-Left Loop Tolerance (ALT_L)	±1.658 % RTP (±0.21 mA)
As-Found Loop Tolerance (AFT_L)	±0.5957 % RTP (±0.076 mA)

The NTSP is conservative with respect to the Analytical Limit and Allowable Value.

TSV Closure Scram, TCV Fast Closure Scram and EOC-RPT Bypass

Calculation JC-Q1C71N652-1, "Instrument Loop Uncertainty and Setpoint Determination for TSV Closure Scram Bypass, TCV Fast Closure Scram Bypass and EOC-RPT Bypass," determines the Nominal Trip Setpoint and Allowable Value for the TSV closure scram bypass; TCV fast closure scram bypass; and the EOC-RPT bypass (referred to as RPS Trip Bypass). The TSV closure scram, TCV fast closure scram and EOC-RPT are automatically enabled when APRM flux increases to the trip reset point. Note that these instrument loops trip on decreasing power and reset on increasing power such that the TSV closure scram, the TCV fast closure scram and EOC-RPT functions are enabled above the reset point.

The RPS Trip Bypass Analytical Limit (≥ 35.4% RTP) is listed in TS SR 3.3.1.1.14. This Analytical Limit is not changed by this plant modification. The RPS Trip Bypass Allowable Value is listed in the Technical Requirements Manual, but is not listed in the TS. The Allowable Value is determined by calculation JC-Q1C71N652-1, which has been revised by this modification to account for the uncertainties associated with PRNMS and the new signal converters. This calculation revision establishes a new Allowable Valve of ≤ 32% RTP and demonstrates that the Nominal Trip Setpoint of 26% RTP is conservative. The Allowable Value in the Technical Requirements Manual is revised to show the new ≤ 32% RTP value.

The assumptions used in calculation JC-Q1C71N652-1 are as follows:

1. All uncertainties are two standard deviations (2σ) unless otherwise specified.

2. M&TE accuracy is assumed to be equal to the larger of the device reference accuracy or the setting tolerance.

3. The APRMs, calibrated weekly, measure average reactor power and provide an unfiltered output referred to as Neutron Flux. The Neutron Flux output is prompt and follows changes in thermal power promptly. For the purposes of this calculation, Neutron Flux will be considered to be equivalent to the RTP.

4. Insulation Resistance Effects are negligible because the design basis event for the loop function does not result in harsh environmental conditions, and the normal environment is relatively mild. Therefore IR = ± 0.0.

5. In these calculations, the Rosemount 510DU Trip Unit "Normal/Normal" Repeatability specification is used as the Reference Accuracy (RA_2), and the "Adverse/Normal" Repeatability specification is used as the trip unit uncertainty (A_2). This is conservative because both of these values from Rosemount include Drift for 6 months in addition to the Reference Accuracy and environmental uncertainties.

   The Normal/Normal Repeatability is ±0.13% Span. The "Adverse/Normal" and "Normal/High" Repeatability is ±0.20% Span.

   Unlike many other instrument loops, the trip units in the 1C71N652 loops are calibrated every refueling outage. A separate drift calculation determined 30 month drift statistics for the Rosemount 510DU Trip Units at GGNS. Therefore, it is not necessary to estimate drift statistics from the Rosemount specifications.

The "Normal/Normal" Repeatability (±0.13% Span) best represents RA_2 because calibrations are performed under stable environmental conditions. The "Adverse/Normal" Repeatability (±0.20% Span) includes expanded environmental limits and is conservatively used for the A_2 uncertainty parameter.

6. The Rosemount Model 510DU is obsolete, and these units may be replaced by the Model 710DU in the future. The specifications for the Model 710DU are equal to or better than the Model 510DU.

7. No Drift is specified by the signal converter vendor. The vendor published Reference Accuracy will be assumed to be applicable as Drift over a period of 6 months.

8. Radiation effects (RE) are negligible and set to zero. Since all of the instruments are located in the Control Room, the design basis for the instrument loops does not result in harsh environmental conditions, and the normal environment is mild, including no exposure to radiation.

9. License Basis Document Change Request (LBDCR) 2011-030 was issued to update EOC RPT and scram bypass Analytical Limits and Allowable Values specified in the TRM and UFSAR per ECN32044 and EC23925. It has been incorporated. The 34.33% RTP allowable value specified in LBDCR 2011-030 is not conservative based on this calculation revision. It has been changed to 32.0% which has been validated to be conservative.

10. No Drift is specified by the vendor for the APRM Analog Isolator. The vendor published Reference Accuracy will be assumed to be applicable as Drift over a period of 6 months.

11. Calibration error is the accuracy to which the APRMs are calibrated to the reactor heat balance. This will be combined with the Neutron Noise Error and accounted for as Process Measurement Accuracy.

The below tables provide the results of calculation JC-Q1C71N652-1.

PARAMETER	INSTRUMENTS A/B/C/D % RTP	PARAMETER	INSTRUMENTS A/B/C/D % RTP
Analytical Limit (AL)	35.4		
Calculated Allowable Value (AV_{CALC})	32.465	Actual Allowable Value (AV)	32.00
Calculated Nominal Trip Reset ($NTRP_{CALC}$)	32.403	Actual Nominal Trip Reset (NTRP)	26.00
		Actual Nominal Trip Setpoint (NTSP)	25.00

PARAMETER	VALUE
Instrument Loops (Circuits)	1C71N652A - D
Loop Uncertainty (LU)	±2.935% RTP
Total Loop Uncertainty (TLU)	±2.997% RTP
Loop Drift (D_L)	±0.6085% RTP
Loop MTE Uncertainty (C_L)	±1.697% RTP
APRM ALT (ALT_1)	±0.132 Volts
Converter ALT (ALT_C)	±0.016 mA
Trip Unit ALT (ALT_2)	±0.021 mA
APRM AFT (AFT_1)	±0.036 Volts
Converter AFT (AFT_C)	±0.036 mA
Trip Unit AFT (AFT_2)	±0.03 mA
As-Left Loop Tolerance (ALT_L)	±1.658 % RTP (±0.21 mA)
As-Found Loop Tolerance (AFT_L)	±0.5957 % RTP (±0.076 mA)

The above calculation revisions indicate that the plant modification maintains conservative margins between Analytical Limits, Allowable Values and the Nominal Trip Setpoints. Response time is improved as reactor power is measured directly from core neutron flux, which changes instantaneously with changes in reactor thermal power, as opposed to indirectly from the first stage turbine pressure which may take several seconds to respond to RTP changes due to steam line process delays. The modification is expected to provide an improvement in accuracy for the determination of the LPSP and HPSP setpoints in terms of reactor power. One of the main reasons underlying this improvement is that the APRM based method uses power directly whereas the TFSP based method infers power from the pressure measurement. Using APRM should result in an improvement in HPSP operating margin since the HPSP is set quite low (~55% RTP) in relation to the Analytical Limit (~70% RTP) because of the need to account for reduced TFSP corresponding to the Analytical Limit due to reduced feedwater heating (100 deg. F). The APRM based method should also provide improved operating margins for the RPS trip bypass setpoint since the error in the power estimated by the APRM method is due primarily to the reactor thermal heat balance error (< 2%), whereas for the TFSP based method the error in power estimate would include this reactor heat balance error and also an additional error due to the turbine heat balance (expected to less accurate than the reactor heat balance).

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50.59

10 CFR 50.59 establishes the conditions under which licensees may make changes to the facility or procedures and conduct tests or experiments without prior NRC approval. Proposed changes, tests and experiments that satisfy the definitions and one or more of the criteria in the rule must be reviewed and approved by the NRC before licensee implementation.

In June 2014, Entergy implemented EC 49880 in accordance with 10 CFR 50.59 that replaced the use of the TFSP instruments with the PRNMS to measure reactor power. On December 9, 2016, the NRC issued NRC Inspection Report 05000416/2016007. In this inspection report, the NRC issued NCV 050000416/2016007-02, when it was identified that Entergy failed to obtain a license amendment prior to implementing a proposed change that resulted in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety. Specifically, modification EC 49880 eliminated the TFSP instruments signals to the Reactor Protection System and replaced the signals with APRM signals which reduced the diversity and resulted in a more than minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component important to safety.

As identified in Section 3.0 above, Entergy letter GNRO-2010/0056 identified that the GGNS instrument setpoint methodology is based on NEDC-31336P-A. Section 3.25 of NEDC-31336P-A specifies that the RPS design purposely chooses TFSP, as opposed to the more direct measurement of power such as neutron flux, in order to assure diversity between the TSV closure and TCV fast closure scram functions and the neutron flux scram function. Section 4.3.2 of Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," indicates that although criterion ii of 10 CFR 50.59 allows minimal increases, licensees must still meet applicable regulatory requirements and other acceptance criteria to which they are committed. Further, Example 6 in Section 4.3.2 indicates NRC approval under this criterion if the change would reduce system/equipment redundancy, diversity, separation or independence. A reduction in diversity would require NRC approval because it would result in more than a minimal increase in the likelihood of occurrence of a malfunction of an SSC important to safety previously evaluated in the updated safety analysis report. As a result of this determination, the proposed change is being submitted as a license amendment request per 10 CFR 50.59(c)(2).

General Design Criteria

The following 10 CFR 50, Appendix A, General Design Criteria are applicable to the permanent plant modification:

> *Criterion 10 - Reactor design.* The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

> *Criterion 12 - Suppression of reactor power oscillations.* The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

> *Criterion 13 - Instrumentation and control.* Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

> *Criterion 15 - Reactor coolant system design.* The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 19 - Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

Criterion 20 – Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

Criterion 21 – Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Criterion 22 – Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Criterion 25 - Protection system requirements for reactivity control malfunctions. The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Criterion 27 - Combined reactivity control systems capability. The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

Criterion 28 - Reactivity limits. The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

Criterion 29 - Protection against anticipated operational occurrences. The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Entergy has evaluated the proposed changes against the applicable regulatory requirements and acceptance criteria and finds the design of the permanent plant modification is consistent with the applicable regulatory criteria described above. The proposed change does not affect compliance with these regulations or guidance and will ensure that the lowest functional capabilities or performance levels of equipment required for safe operation are met. The technical analysis in Section 3.0, above, concludes that the proposed changes for the permanent replacement of the TFSP output signals with the PRNMS output signals continue to assure that the design requirements and acceptance criteria are met.

4.2 No Significant Hazards Consideration Analysis

In accordance with the requirements of 10 CR 50.90, "Application for amendment of license, construction permit, or early site permit," Entergy Operations, Inc. (Entergy) requests an amendment to Renewed Facility Operating License NPF-29 for the Grand Gulf Nuclear Station (GGNS). The proposed amendment revises the Updated Final Safety Analysis Report (UFSAR) descriptions for the replacement of the Turbine First Stage Pressure (TFSP) output signals with Power Range Neutron Monitoring System (PRNMS) output signals.

Entergy Operations, Inc. (Entergy) has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. **Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?**

Response: No.

The proposed modification does not result in a change to the safety related functions including Low Power Setpoint (LPSP) and High Power Setpoint (HPSP), Turbine Stop Valve (TSV) closure and Turbine Control Valve (TCV) fast closure scram enable/bypass, and End of Cycle Recirculation Pump Trip (EOC-RPT) enable/bypass. The accidents potentially affected by the TFSP instrumentation are the turbine trip event (UFSAR Section 15.2.3), generator load rejection event (UFSAR Section 15.2.2), control rod drop accident (UFSAR Section 15.4.9) and rod withdrawal error (UFSAR Section 15.4.1). The proposed use of PRNMS signal outputs as inputs to the trip units will maintain the safety related functions credited in the evaluated events. Furthermore, the proposed modification makes no changes to the existing PRNM system inputs, system software or hardware architecture.

Overall protection system performance will remain within the bounds of the previously performed accident analyses since the proposed modification does not change the Reactor Protection System (RPS) or the Rod Control and Information System (RCIS). The same RPS and RCIS instrumentation will continue to be used. The protection systems will continue to function in a manner consistent with the plant design basis. The proposed modifications will not adversely affect accident initiators or precursors nor adversely alter the design assumptions and conditions of the facility or the manner in which the plant is operated and maintained with respect to such initiators or precursors.

The proposed modification will not prevent the capability of structures, systems, and components (SSCs) to perform their intended functions for mitigating the consequences of an accident and meeting applicable acceptance limits.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The use of PRNMS for determining reactor power will ensure that the protective functions EOC-RPT, TSV closure and TCV fast closure direct scram functions, and the rod pattern controller (RPC) and Rod Withdrawal Limiter functions credited in the safety analyses are maintained. With these automatic functions maintained, the proposed modification does not create the possibility of a new or different kind of accident from any accident previously evaluated in the UFSAR.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures will be introduced as a result of the proposed modification. No new or different accidents result from the proposed modification. The proposed modification will not alter the performance of the RPS, RCIS and PRNMS.

Thus, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed modification does not alter the manner in which safety limits, safety setpoints, or limiting conditions for operation are determined. The PRNMS hardware and software are not changed by this modification. The modified system responds to a loss of power, and a restoration of power, in the same way as the TFSP system would have responded. The proposed modification makes no changes to the PRNMS, RPS or RCIS human-system interfaces. The equipment credited to perform a safety function has been designed and installed to the applicable quality standards and maintained the required redundancy. The proposed modification is expected to provide an improvement in accuracy for the determination of the low power setpoint and high power setpoint in terms of reactor power. The replacement of the TFSP output signals with the PRNMS output signals does not reduce the diversity of the RPS trip functions by use of a more direct measurement of power given the additional diverse capabilities available. The proposed modification maintains conservative margins between Analytical Limits, Allowable Values and the Nominal Trip Setpoints.

The proposed change does not impact accident offsite dose, containment pressure or temperature, Emergency Core Cooling System settings, Reactor Core Isolation Cooling System settings or RPS settings, or other parameter that could affect a margin of safety.

Therefore, the proposed amendment does not involve a significant reduction in a margin of safety.

Based on the above, Entergy concludes that the proposed change presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.3 Conclusions

In conclusion, based on the considerations discussed above, 1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, 2) such activities will be conducted in compliance with the Commission's regulations, and 3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, and would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

6.0 REFERENCES

6.1. Letter from A. Wang, USNRC, to Entergy Operations, Inc., "Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment RE: Power Range Neutron Monitoring System Replacement (TAC No. ME2531)," March 28, 2012. (ADAMS Accession Number ML120400319) [Amendment No. 188]

6.2 Letter from T. R. Farnholtz, USNRC, to Entergy Operations, Inc., "Grand Gulf Nuclear Station – NRC Evaluations of Changes, Tests, and Experiments and Permanent Plant Modifications Baseline Inspection Report 05000416/2016007," December 9, 2016. (ADAMS Accession Number ML16348A222)

6.3 GEH Nuclear Energy Report 004N6431, Revision 1, "Technical Justification of the Grand Gulf Nuclear Station Modification to Operational Bypass Signal, Replacing Turbine First Stage Pressure with APRM Neutron Flux," January 2018.

6.4 Letter GNRO-2010/0056 from M. A. Krupa, Entergy Operations, Inc. to USNRC, "License Amendment Request Extended Power Uprate," September 8, 2010. (ADAMS Accession No. ML102660403)

6.5 Instrument Society of America (ISA) Standard 67.04 Part II, 1994, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."

6.6 NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," September 1996. (ADAMS Accession No. ML072950103)

6.7 Letter from A. Wang, USNRC, to Entergy Operations, Inc., "Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment RE: Extended Power Uprate (TAC No. ME4679)," July 18, 2012. (ADAMS Accession Number ML121210020) [Amendment No. 191]

6.8 GE Nuclear Energy, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," NEDC-32410P-A Supplement 1, November 1997.

6.9 Letter GNRO-2009-00054 from M. A. Krupa, Entergy Operation, Inc., to USNRC, "License Amendment Request Power Range Neutron Monitoring System Upgrade," November 3, 2009. (ADAMS Accession Number ML093140463)

6.10 USNRC, Branch Technical Position (BTP) 7-19, "Guidance for Evaluation of Diversity and Defense in Depth in Digital Computer-based Instrumentation and Control Systems," July, 2012. (ADAMS Accession Number ML110550791)

6.11 Letter GNRO-2011/00039 from M. A. Krupa, Entergy Operation, Inc., to USNRC, "Responses to NRC Requests for Additional Information Pertaining to License Amendment Request for Power Range Neutron Monitoring System (TAC No. ME2531)," May 26, 2011. (ADAMS Accession Number ML111460590)

6.12 Letter GNRO-2013/00012 from B. S. Ford, Entergy Operations, Inc., to USNRC, "Maximum Extended Load Line Limit Analysis Plus (MELLLA+) License Amendment Request," September 25, 2013. (ADAMS Accession Number ML13269A140)

6.13 Letter GNRO-2013/00100 from K. Mulligan, Entergy Operations, Inc., to USNRC, "Maximum Extended Load Line Limit Analysis Plus (MELLLA+) License Amendment Request - Responses to Requests for Supplemental Information," December 30, 2013. (ADAMS Accession Number ML13364A286)

6.14 Letter from A. Wang, USNRC, to Entergy Operations, Inc., "Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment Regarding Maximum Extended Load Line Limit Analysis Plus (TAC No. MF2798)," August 31, 2015. (ADAMS Accession Number ML15229A219) [Amendment No. 205]

Attachment 2 to GNRO-2018/00012

Updated Final Safety Analysis Report Changes (Mark-up)

change position from that of normal plant operation. If these valves for any reason change position from that of normal operation, they will not have any adverse affect on accident mitigation or normal shutdown efforts. As such, valves in this category do not have any safety function other than maintaining system integrity.)

7.1.2.7 Safety System Settings

The safety system set points are listed in the design basis discussions for each safety system. The settings are determined based on operating experience and conservative analyses. The settings are high enough to preclude inadvertent initiation of the safety action, but low enough to assure that significant margin is maintained between the actual setting and the limiting safety system settings. Instrument drift, ease of set point adjustment, and repeatability are considered in the set point determination. The margin between the limiting safety system settings and the actual safety limits include consideration of the maximum credible transient in the process being measured.

INSERT 1

The method employed to establish adequate margins for instrument set point drift, inaccuracy, and calibration uncertainty as discussed in NRC Regulatory Guide 1.105 is explained by reference to Figure 7.1-10. Because of the generic nature of this figure it is not drawn to any scale and is used solely to illustrate the qualitative relationships of the various margins. Starting with a Safety Limit as indicated at the extreme right hand of the figure, the first margin extends to the point marked Analytical Limit. This margin is there to account for uncertainties in the calculational model used but excludes allowances for instrumentation. Thus the calculational model can assume ideal or perfect instruments. The next margin is between the Analytical Limit and the Allowable Value of the parametric set point, and accounts for instrument errors and calibration capability for the specific instrumentation. The remaining margin which is of interest from a safety standpoint is that shown between the Allowable Value and the Instrument Set Point. This margin is that which is deemed adequate to cover instrument drift which might occur during the established surveillance period. It follows that if during the surveillance period an instrument has drifted from its set point in a nonconservative direction but not beyond the allowable value, then the instrument performance is still within the requirements of the plant safety analysis.

INSERT 1

Entergy Letter GNRO-2010/0056 identified that the instrument setpoint methodology currently implemented at GGNS is based on Instrument Society of America (ISA) Standard 67.04 Part II, 1994, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," and the General Electric Hitachi (GEH) Instrument Setpoint Methodology (ISM) specified in NEDC-31336P-A, "General Electric Instrument Setpoint Methodology." Additionally, in the NRC safety evaluation for Amendment No. 191, the NRC states: "Entergy stated that its setpoint calculations in support of the EPU LAR were based on NEDC-31336P-A, "General Electric Instrument Setpoint Methodology," September 1996 (Reference 120), which includes the NRC-approved SE dated November 6, 1995." GEH NEDC-31336 P-A, Section 3.25, states, in part:

"Turbine first stage pressure has been historically used as the parameter to approximate reactor power and effect the actual trip bypass. The Reactor Protection System (RPS) design purposely chooses this parameter, as opposed to the more direct measurement of power such as neutron flux, in order to assure diversity between the TSVC [turbine stop valve closure] and TCVFC [turbine control valve fast closure] scram functions and the neutron flux scram function."

This statement is not completely applicable as it pertains to GGNS because the UFSAR (Section 7.2.1.1.4.2 paragraph d and e) does not credit the neutron flux scram for diversity from the TSVC and TCVFC scrams. Rather, it credits the reactor vessel high-pressure trip signal. As such, Section 3.25 of NEDC-31336 P-A is considered not applicable based on the current design and licensing basis for the GGNS.

All manual bypass switches are in the control room, under the direct control of the control room operator. The bypass status of trip system components is continuously indicated in the control room.

7.2.1.1.4.4.1 Neutron Monitoring System

Bypasses for the neutron monitoring system channels are described below.

The neutron monitoring scram sensor trip contacts for IRM and APRM can be bypassed by hand operated selector switches located on the reactor control benchboard in the control room. A single APRM channel (1, 2, 3, or 4) may be bypassed via the APRM bypass switch, which is an optical joystick.

Bypassing IRM channels is controlled by two selector switches. One switch controls channels A, C, E, and G while the second switch controls channels B, D, F, and H. Each selector switch can bypass only one channel at a time.

Bypassing either an APRM or an IRM channel will not inhibit the neutron monitoring system from providing protective action when required.

The NMS operating bypasses are controlled by the reactor mode switch located on the operator control console in the control room. When the reactor mode switch is in the "RUN" mode, the IRM trips are bypassed; protection is provided by the APRM trips. Refer to the Technical Specifications for NMS trips.

7.2.1.1.4.4.2 Turbine Stop Valve and Turbine Control Valve Fast Closure

the Neutron Monitoring System

The turbine control valve fast closure scram and turbine stop valve closure scram are automatically bypassed if reactor power is low (below the bypass setpoint), as indicated by ~~turbine first stage pressure~~. Closure of these valves below a low initial power level does not threaten the integrity of any radioactive material release barrier. Turbine control valve fast closure and turbine stop valve closure trip bypass is effected by ~~four pressure transmitters associated with the turbine first stage~~. Any one channel in a bypass state produces a control room annunciation. No single failure of a transmitter can prevent a turbine stop valve closure scram or turbine control valve fast closure scram. In addition, this bypass is automatically removed when ~~the~~

Replace with Insert 2

INSERT 2

four independent reactor power signals associated with the four divisions of the Power Range Neutron Monitoring System.

reactor power

~~turbine first stage pressure~~ exceeds the bypass reset point. This reset set point established to ensure the bypass is removed prior to reactor power exceeding the Analytic Limit 35.4 percent of rated power.

~~Turbine first stage pressure is sensed from two physically separate and redundant pressure taps. Each pressure tap is piped to two pressure transmitters, which sense first stage pressure~~. ~~Redundancy has been achieved by connecting one pressure transmitter output in parallel with each of the turbine stop valve and turbine control valve fast closure trip contacts in each of four scram trip logics~~.

Replace with Insert 3

7.2.1.1.4.4.3 Main Steam Isolation Valves

At plant shutdown and during initial plant startup, a bypass is required for the main steam line isolation valve closure scram trip in order to properly reset the reactor protection system. This bypass has been designed to be in effect when reactor pressure is less than normal reactor operating pressure and the mode switch is in the SHUTDOWN, REFUEL, or STARTUP position. The bypass allows plant operation when the main steam line isolation valves are closed during low power operation. The bypass is removed when the mode switch is placed in the RUN position.

7.2.1.1.4.4.4 Scram Discharge Volume Level

The scram discharge high-water-level trip bypass is controlled by the manual operation of four keylocked bypass switches, one for each channel, and the keylocked mode switch. The mode switch must be in the SHUTDOWN or REFUEL position. Four bypass channels emanate from the four banks of the RPS mode switch and are connected into the RPS logic. This bypass allows the operator to reset the reactor protection system scram relays so that the system is restored to operation allowing the operator to drain the scram discharge volume. Resetting the trip actuators opens the scram discharge volume vent and drain valves. An annunciator in the main control room indicates the bypass condition.

7.2.1.1.4.4.5 Mode Switch in Shutdown

The scram initiated by placing the mode switch in SHUTDOWN is automatically bypassed after a short time delay. The bypass allows the control rod drive hydraulic system valve lineup to be restored to normal. An annunciator in the control room indicates the bypassed condition.

INSERT 3

Reactor power is sensed by four physically separate and independent divisions of the Power Range Neutron Monitoring System. Redundancy has been achieved by connecting one reactor power output signal in parallel with each of the turbine stop valve and turbine control valve fast closure trip contacts in each of the four scram trip logics.

the float type level switch sensing instrumentation is redundant to, and diverse (manufacture, design logic and operating principle) from the level transmitter sensing instrumentation. Separate instrument lines and tap locations are used for the level switches and transmitters.

RPS manual controls also comply with the single failure criterion. Four manual scram pushbuttons are arranged into two groups on the operator's control console, and are separated by approximately 6 inches within each group to permit the operator to initiate manual scram with one motion of one hand. The two groups of manual scram pushbuttons are separated by approximately 3 feet, and the switch contact blocks are enclosed within metal barriers.

The mode switch consists of a single manual actuator shaft with four distinct, steel barrier-separated, switch banks. Each bank is housed within a fire retardant cover. Contacts from each bank are wired in conduit to individual metallic terminal boxes.

The scram discharge volume high-water-level trip bypass requires manual operation of the mode switch and one of four bypass switches for each trip channel. Each of these four bypass switches, in conjunction with a set of mode switch contacts, is used to energize the corresponding channel bypass relays to establish the trip bypass. There is no single failure of this bypass function that will satisfy the condition necessary to establish the bypass condition. Hence, the function complies with the single-failure criterion.

The main steam line valve closure trip operating bypass is implemented with redundant mode switch contacts in a similar manner.

The turbine stop valve closure trip and control valve fast closure trip operating bypass complies with the single-failure criterion. ~~Two pressure transmitters are mounted on each of two turbine first stage pressure taps. Wiring from the pressure transmitters is routed in conduit to the RPS cabinets in the control room.~~ The logic configuration for the bypass is the standard one-out-of-two twice arrangement such that a single bypass is associated with a single trip logic for stop valve closure and a single trip logic for control valve fast closure.

Replace with Insert 4

INSERT 4

Reactor power is sensed by four physically separate and independent divisions of the Power Range Neutron Monitoring System (PRNMS). Wiring from the PRNMS is routed to the RPS cabinets and divisional separation is maintained along the route.

Loss of hydraulic pressure in the EHC oil lines which initiates fast closure of the control valves is monitored. These measurements provide indication that fast closure of the control valves is imminent.

This measurement is felt to be adequate and a proper variable for the protective function taking into consideration the reliability of the chosen sensors relative to other available sensors and the difficulty in making direct measurements of control-valve fast closure rate.

Since the mode switch is used to connect appropriate trip relays into the RPS logic depending upon the operating state of the reactor, the selection of particular contacts to perform this logic operation is an appropriate means for obtaining the desired function.

PRNMS reactor power

Since the intent of the turbine stop valve closure trip and control valve fast closure trip operating bypass is to permit continued reactor operation at low power levels when the turbine stop or control valves are closed, the selection of ~~turbine first stage pressure~~ is an appropriate variable for this bypass function. ~~In the power range of reactor operation, turbine first stage pressure is essentially linear with increasing reactor power. Consequently, this variable provides the desired measurement of power level.~~

Due to the manual action required for scram discharge volume high-water-level trip bypass, this design requirement is satisfied by operator interaction with a single bypass switch and the mode switch.

7.2.2.1.2.3.1.9 Capability for Sensor Checks (IEEE Std. 279-1971, paragraph 4.9)

During reactor operation, the analog display of each of the four redundant sensor channels for the following RPS trip variables may be directly compared:

a. Scram discharge volume high-water-level

b. Reactor vessel low- and high-water-level

c. Drywell high-pressure

d. Reactor vessel high-pressure

The APRMs are calibrated to reactor power by using a reactor heat balance and the TIP system to establish the relative local flux profile. LPRM gain settings are determined from the local flux profiles measured by the TIP system once the total reactor heat balance has been determined.

The gain-adjustment-factors for the LPRMs are produced as a result of the process computer nuclear calculations involving the reactor heat balance and the TIP flux distributions. These adjustments, when incorporated into the LPRMs permit the nuclear calculations to be completed for the next operating interval and establish the APRM calibration relative to reactor power. The APRM gains are adjusted using the instrument's front panel display or accepting the APRM gain calculated from the percent core thermal power (% CTP) downloaded from the Plant Process Computer.

During reactor operation, one manual scram pushbutton may be armed and depressed to test the proper operation of the switch, and once the RPS has been reset, the other switches may be armed and depressed to test their operation one at a time. For each such operation, a control room annunciation will be initiated and the process computer will print the identification of the pertinent trip.

Operation of the reactor system mode switch from one position to another may be employed to confirm certain aspects of the RPS trip logics during periodic test and calibration. During tests of the trip logics, proper operation of the mode switch contacts may be easily verified by noting that certain trip relays are connected into the RPS logic and that any other trip relays are disconnected from the RPS logic in an appropriate manner of the given position of the mode switch.

In the startup and run modes of plant operation, procedures are used to confirm that scram discharge volume high-water-level sensor channels are not bypassed as a result of operating the bypass switch. In the shutdown and refuel modes of plant operation, a similar procedure is used to confirm that all four sensor channels are bypassed. Due to the ON-OFF nature of the bypass function, calibration is not meaningful.

reactor power

Administrative control must be exercised to place one ~~turbine first stage pressure~~ trip unit in the calibration mode for the periodic test. During this test, a variable calibration signal may be introduced to operate the trip unit at the set point value. When the condition for bypass has been achieved on an individual sensor under test, the control room annunciator for this bypass function will be initiated. If the RPS trip logic associated with this sensor had been in its tripped state, the process computer will log the return to the normal state for the RPS trip logic. When the plant is operating above 35.4 percent of rated power, testing of the turbine stop valve and control valve fast closure trip channels will confirm that the bypass function is not in effect.

Operation of the reset switch following a trip of one RPS trip system will confirm that the switch is performing its intended function. Operation of the reset switch following scram will confirm that all portions of the switch and relay logic are functioning properly since half of the control rods are returned to a normal state for one actuation of the switch.

A manual scram switch permits each individual trip logic, trip system, and trip actuator to be tested on a periodic basis. Testing of each process sensor of the protection system also affords an opportunity to verify proper operation of these components.

7.2.2.1.2.3.1.11 Channel Bypass or Removal from Operation (IEEE Std. 279-1971, paragraph 4.11)

The following RPS trip variable has no provision for channel bypass or removal from service because of the use of valve position limit switches as the channel sensor:

a. Main steam line isolation valve closure trip

During periodic test of any one sensor channel, a transmitter may be valved out of service and returned to service under administrative control procedures. Since only one transmitter is valved out of service at any given time during the test interval, protective capability for the following RPS trip variables is maintained through the remaining instrument channels:

b. b.Turbine stop valve closure trip

c. c.Scram discharge volume high-water-level trip

d. d.Turbine control valve fast closure trip

TABLE 7.2-1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SPECIFICATIONS (Continued)

Bypass Function	Instrument	Range(1)	Sensor Channels Provided (2)
Reactor vessel high water level	Level transmitter (3)	-160/0/60" HO	4
Scram discharge volume high water level trip bypass	Manual switch	N/A	4
Turbine stop valve and control valve fast closure trip bypass	~~Pressure transmitter~~ Reactor Power (4)	0 to ~~100%~~ 125% Power	4
Main steam line isolation valve trip bypass	Manual switch	N/A	4

(1) See Technical Specifications/Technical Requirements Manual (TRM) for operational limits, levels requiring protective action, accuracy, trip settings, margin between operational limits, and response time requirements.

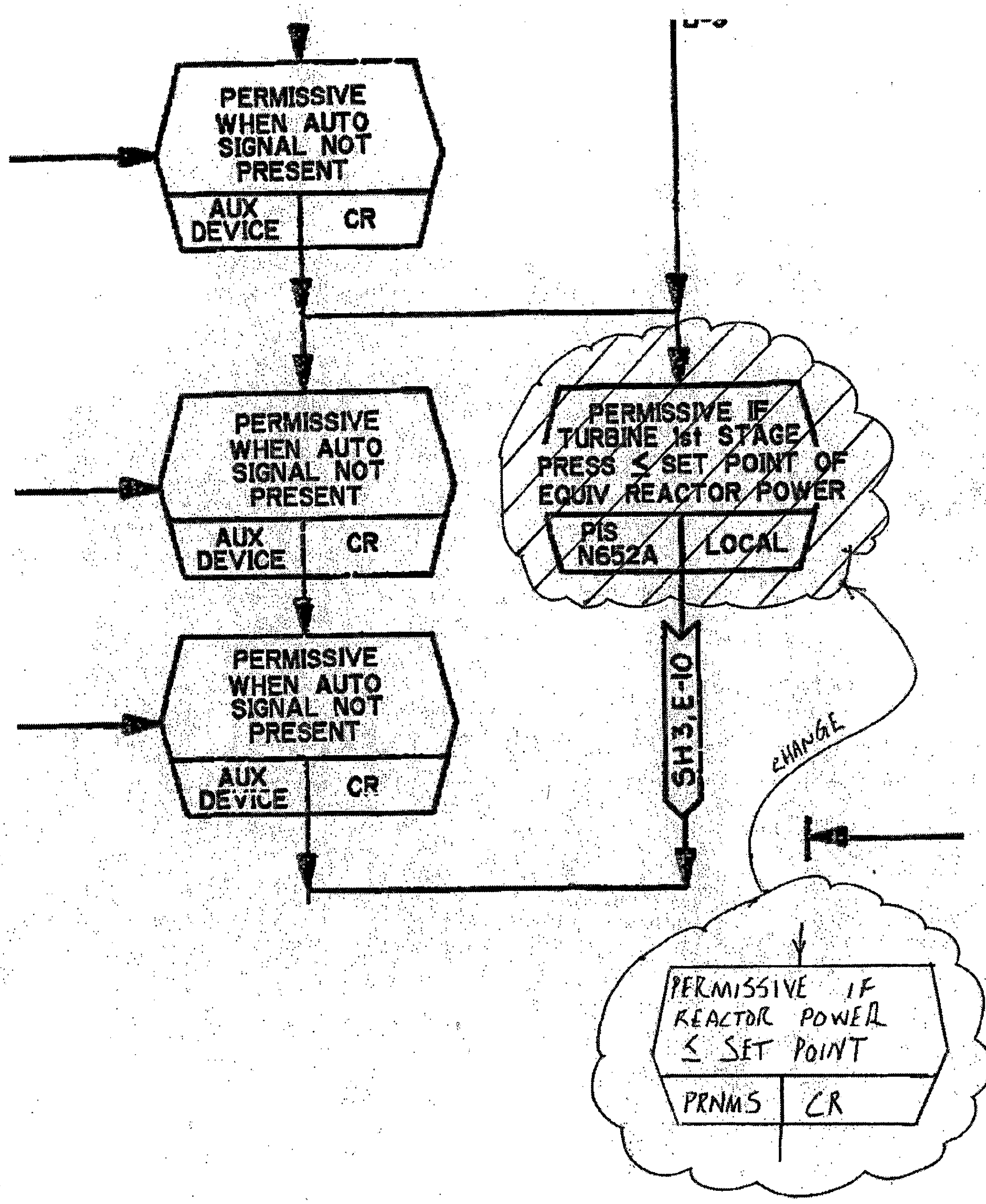
(2) See Technical Specifications/Technical Requirements Manual (TRM) for the minimum number of channels required.

(3) A common level transmitter is used for both high and low reactor vessel trips - separate trip units monitor the common level signal.

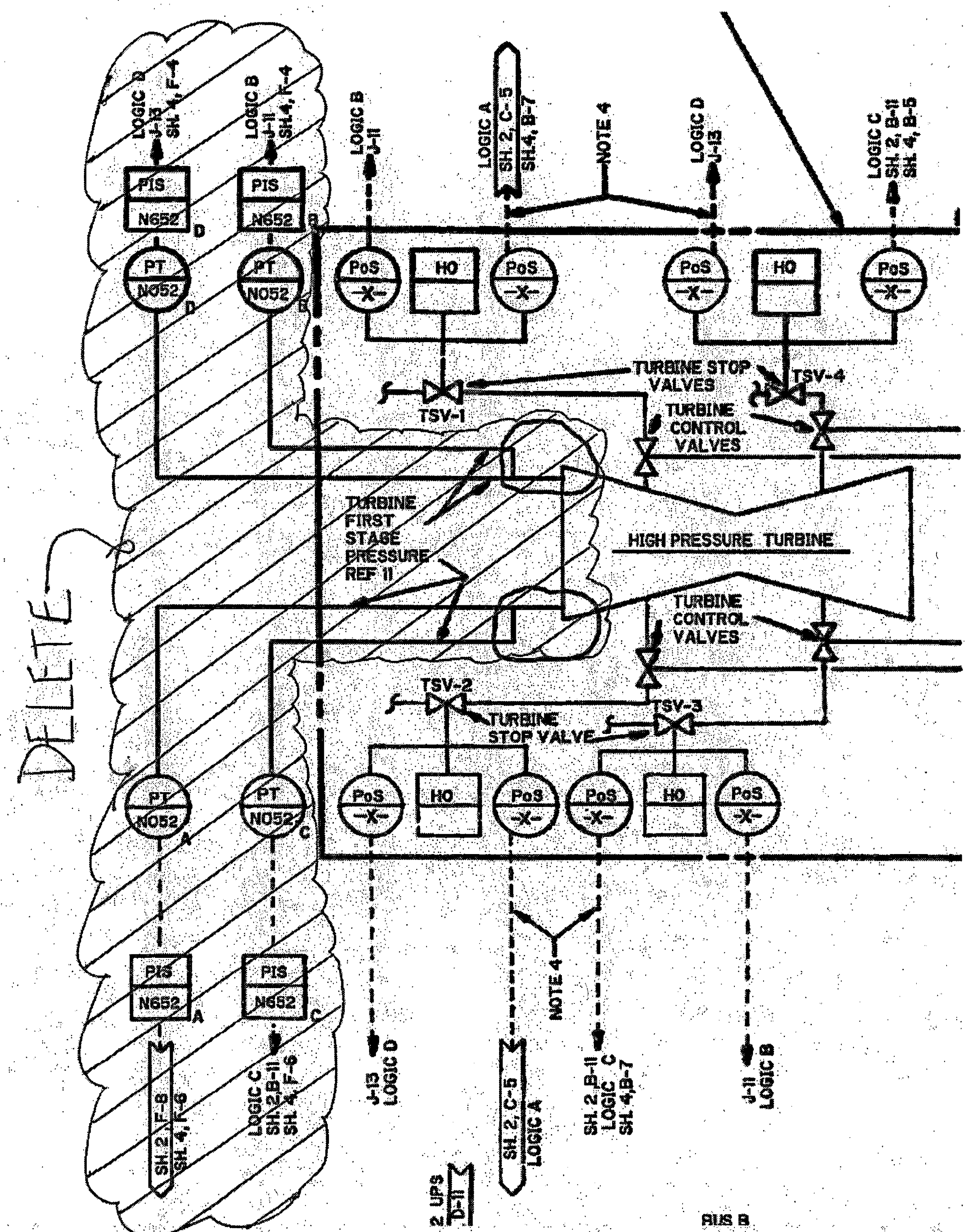
(4) ~~Deleted~~ Reactor power signals derived from the Power Range Neutron Monitoring System.

(5) This is a mechanical setting and can be adjusted over the full range, 90% to 100%.

PARTIAL VIEW OF UFSAR FIGURE 7.2-001B (ZONE E-8)



PARTIAL VIEW OF FIGURE 7.2-001C (ZONE B-9)



directly from the Power Range Neutron Monitoring System.

When the LPSP is reached, the rod pattern controller no longer inhibits control rod motion. Should a rod drop accident occur at power levels above LPSP, the reactivity change would have less effect than if it occurred at power levels below LPSP. This power level is derived ~~by measuring first stage turbine pressure using transmitters and alarm units.~~ There are two channels of instruments which are redundant and separated divisionally. These trip functions are input to the proper rod activity control cabinet and both instrument channels must trip to bypass the RPCS. These instruments are continuously monitored, and any instruments out of service or gross failure is alarmed and indicated in the control room.

From the LPSP to the HPSP, rod withdrawals are restricted to prevent excessive changes in the heat flux rate. From the HPSP to 100-percent power, rod withdrawals are further restricted to prevent excessive change in the heat flux rate. A fixed number of notches is allowed for rod movement, and motion beyond this point is blocked.

Shutdown follows the same rules as above but in reverse. The only difference is that an approach alarm, called the low-power alarm point, is provided so that the operator may prepare valid rod positions for proper shutdown below the LPSP.

Implementing EPU resulted in rescaling the lower bound of the LPSP to maintain the AL in terms of absolute power. The upper bound AL for LPSP was not rescaled. Additionally, because the high pressure turbine was modified to support EPU power levels, new allowable values (AVs) were established for both the upper and lower bounds of LPSP in units of psig.

7.6.1.7.3.1 Bypass of the RPCS

Because of the possibility of stuck rods, provisions are made to bypass failed inputs per the following rules. Substitute rod positions may be entered into the RPCS providing:

a. Only one entry per channel per subgroup is allowed.

b. The same position cannot be entered into both channels.

c. Upon rod motion and a new position scan, the substitute rod position will be overlayed with new data.

7.6.1.8.1.1 Safety Classification

The recirculation pump trip (RPT) system is a nuclear safety-related (class IE) system. The initiation signals for low frequency MG start are non-class IE and are isolated from the class IE circuits.

7.6.1.8.1.2 Circuit Sharing

Sensors and logic circuitry are shared with RPS.

7.6.1.8.2 Power Sources

The RPT system utilizes two types of power from the same sources as the reactor protection system (RPS); 120 V ac from nonessential RPS motor-generators is supplied for the sensor channels and essential 125 V dc from station batteries is supplied for the logic trip circuits for RPT.

7.6.1.8.3 Equipment Design

7.6.1.8.3.1 Initiating Circuits

RPS inputs sense turbine stop valve closure (turbine trip) or turbine control valve fast closure (load rejection). These inputs utilize four-division RPS logic and are combined into the two-divisional two-out-of-two systems utilized for RPT function. The devices utilized to sense turbine trip and full load rejection are discussed in subsection 7.2.1.1.4.2. Figure 7.2-7 is typical of the RPT initiation circuitry.

7.6.1.8.3.2 Logic

[HISTORICAL INFORMATION] [The basic logic arrangement is shown on Figure 7.2-3. It is a two-divisional two-out-of-two design for the turbine control valve and two-out-of-two for the turbine stop valve. It receives signals from each of four RPS divisions. Initiation requires confirmation by sensors located in two or more RPS divisions. Failure to initiate requires failure in more than two RPS divisions. Inputs per division are combined in two-out-of-two configurations.]

Each RPT division causes both recirculation pumps to trip off the main power supply.

RPT is automatically bypassed if reactor power is less than 40 percent of its rated value as indicated by ~~turbine first stage pressure.~~ the Power Range Neutron Monitoring System. No single failure in the bypass circuit can prevent an RPT trip.

7.6.1.8.3.3 Actuated Devices

The output from the trip system allows current to flow into the breaker trip coils when a trip signal is received. The breakers interrupt the main power supply when the coil is energized.

7.6.1.8.3.4 Separation

Sensors utilized to monitor for turbine trip and full load rejection are incorporated in the reactor protection system, where they are combined into a two-divisional system for input to the RPT system. All system wiring outside the cabinets is run in accordance with applicable separation requirements. Cables from sensors and power cables are routed such that no single event involving a single panel, cabinet, or raceway can disable the RPT function.

7.6.1.8.3.5 Testability

See subsection 7.2.1.1.4.8.

7.6.1.8.4 Environmental Considerations

The electrical modules and sensors are located in the control room and/or the turbine building. The environmental conditions for these areas are shown in Section 3.11.

7.6.1.8.5 Operational Considerations

7.6.1.8.5.1 General Information

Trip logic is designated by divisions A, B, C, and D and actuation devices (breaker trip coil) by divisions 1 and 2. The trip conditions of sensors and logic devices are shown in Figure 7.2-1 (RPS IED).

7.6.1.8.5.2 Operator Information

a. Indicators

Turbine controls and valves are designed so that the turbine stop and control valves will close upon loss of complete system power or hydraulic pressure.

7.7.1.5.3.4.3 Turbine Generator to Reactor Protection System Interface

Two conditions which initiate reactor scram are turbine stop valve closure and turbine control valve fast closure when reactor power is above a preselected percent of rated power. (See subsection 7.2.1.1.4.4.2.)

The turbine stop valve closure signal is generated before the turbine stop valves have closed more than 10 percent. This signal originates from pressure transmitters and trip units which sense hydraulic trip fluid pressure decay which is indicative of stop valve motion away from fully open. Two pressure transmitters and trip units are provided for each turbine stop valve. The pressure transmitters and trip units are electrically isolated from each other and from other turbine plant equipment.

The control valve fast closure signal is monitored by the turbine control fluid pressure transmitters and trip units which sense control fluid pressure decay which is indicative of fast control valve closure. These transmitters provide the RPS inputs within 30 milliseconds after the control valves start to close in a fast closure mode.

Power Range Neutron Monitoring System reactor power signals

~~Four turbine first stage pressure transmitters and trip units, which measure equivalent steam flow,~~ are provided for bypassing the stop valve closure and control valve fast closure inputs at low power levels.

7.7.1.5.3.4.4 Turbine-Generator to Containment and Reactor Vessel Isolation Control System Interface

7.7.1.5.3.4.4.1 Main Condenser Vacuum Switches

There are four independent main condenser vacuum transmitters and trip units for the purpose of providing an isolation signal to the NSSS main steam isolation valves. Each vacuum sensor has its own isolation (root) valve. The trip units actuate on low vacuum. The trip unit setting is selected so that it is compatible with safe turbine and main condenser operating and design conditions should loss of vacuum occur. Condenser vacuum transmitters and trip units are also discussed in subsection 7.3.1.1.2.4.1.13.

15.2.3.2.1.2 Turbine Trip w/o Bypass

Turbine trip at high power w/o bypass produces the sequence of events listed in Table 15.2-5A.

15.2.3.2.1.3 Deleted

15.2.3.2.2 Systems Operation

15.2.3.2.2.1 Turbine Trip with Bypass

All plant control systems maintain normal operation unless specifically designated to the contrary.

Turbine stop valve closure initiates a reactor scram trip and recirculation pump trip via turbine stop valve trip fluid pressure signals for power levels greater than 35.4 percent NBR. Credit is taken for successful operation of the reactor protection system.

Turbine stop valve closure initiates recirculation pump trip thereby terminating the jet pump drive flow.

The pressure relief system which operates the relief valves independently when system pressure exceeds relief valve instrumentation set points is assumed to function normally during the time period analyzed.

15.2.3.2.2.2 Turbine Trip w/o Bypass

Same as subsection 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed for the entire transient time period analyzed.

15.2.3.2.2.3 Turbine Trip at Low Power with w/o Bypass

Same as subsection 15.2.3.2.2.1 except that failure of the main turbine bypass system is assumed.

It should be noted that below 35.4 percent NB rated power level, a main stop valve scram trip inhibit signal derived from the ~~first stage pressure of the turbine~~ is assumed to be activated. This is done to eliminate the stop valve scram trip signal from scramming the reactor provided the bypass system functions properly. In other words, the bypass would be sufficient at this low power to accommodate a turbine trip without the necessity of

Power Range Neutron Monitoring System

A reactor scram is initiated when the stop valves trip fluid pressure decays, and the signal is present before the stop valves start to close. This signal originates from pressure transmitters and trip units which sense hydraulic trip fluid pressure decay which is indicative of stop valve motion away from fully open.

This stop valve scram trip signal is assumed to be automatically bypassed when the reactor is below 35.4 percent NB rated power level.

Reduction in core recirculation flow is initiated by the trip units associated with the main stop valves, which actuate trip circuitry which trips the recirculation pumps.

15.2.3.3.3 Results

15.2.3.3.3.1 Turbine Trip with Bypass

A turbine trip with the bypass system operating normally is simulated at 105 percent of the initially licensed NB rated steam flow conditions in Figure 15.2-4 for the initial cycle.

Neutron flux increases rapidly because of the void reduction caused by the pressure increase. However, the flux increase is limited to 111 percent of rated by the stop valve scram and the RPT system. Peak fuel surface heat flux does not exceed its initial value. MCPR for the transient does not change significantly.

15.2.3.3.3.2 Turbine Trip w/o Bypass

The results for a turbine trip w/o bypass at 100% power and 105% rated steam flow are presented in Reference 16. The peak neutron flux is limited to 162% of rated by the reactor scram and the peak fuel surface heat flux does not exceed 101% of its initial value. The MCPR for this transient remains above the safety limit for incidents of moderate frequency and, therefore, the design basis is satisfied.

15.2.3.3.3.3 Turbine Trip w/o Bypass, Low Power

Below 35.4 percent of rated power, the turbine stop valve closure and turbine control valve closure scrams are assumed to be automatically bypassed. At these lower power levels, ~~turbine first stage pressure~~ is used to initiate the scram logic bypass. The scram which terminates the transient is

the Power Range Neutron Monitoring System initiated by high vessel pressure. The bypass valves are assumed to fail; therefore, system pressure will increase until the pressure relief set points are reached. At this time, because of the relatively low power of this transient event, relatively few relief valves will open to limit reactor pressure. Peak pressures are not expected to greatly exceed the pressure relief valve set points and will be significantly below the RCPB transient limit of 1375 psig. Peak surface heat flux and peak fuel center temperature remain at relatively low values and MCPR is expected to remain well above the GETAB safety limit.

15.2.3.3.4 Considerations of Uncertainties

Uncertainties in these analyses involve protection system settings, system capacities, and system response characteristics. In all cases, the most conservative values are used in the analyses. For example:

a. Slowest allowable control rod scram motion is assumed.

b. Scram worth shape for all-rod-out conditions is assumed.

c. Minimum specified valve capacities are utilized for overpressure protection.

d. Set points of the safety/relief valves include errors (high) for all valves.

15.2.3.4 Barrier Performance

15.2.3.4.1 Turbine Trip with Bypass

For the initial cycle, peak pressure in the bottom of the vessel reaches 1161 psig which is below the ASME Code limit of 1375 psig for the reactor cooling pressure boundary. Vessel dome pressure does not exceed 1154 psig. The severity of turbine trips from lower initial power levels decreases to the point where a scram can be avoided if auxiliary power is available from an external source and the power level is within the bypass capability.

15.2.3.4.2 Turbine Trip w/o Bypass

The safety/relief valves open and close sequentially as the stored energy is dissipated and the pressure falls below the set points of the valves. Peak nuclear system pressure reaches 1231 psig for this event at 100% power and 105% flow. Peak dome

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8.a, b. Scram Discharge Volume Water Level-High (continued)

in MODES 1 and 2, and in MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies, since these are the MODES and other specified conditions when control rods are withdrawn. At all other times, this Function may be bypassed.

9. Turbine Stop Valve Closure, Trip Oil Pressure-Low

Closure of the TSVs results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, a reactor scram is initiated at the start of TSV closure in anticipation of the transients that would result from the closure of these valves. The Turbine Stop Valve Closure, Trip Oil Pressure-Low Function is the primary scram signal for the turbine trip event analyzed in Reference 4. For this event, the reactor scram reduces the amount of energy required to be absorbed and, along with the actions of the End of Cycle Recirculation Pump Trip (EOC-RPT) System, ensures that the MCPR SL is not exceeded.

REACTOR POWER SIGNALS DERIVED FROM THE POWER RANGE NEUTRON MONITORING SYSTEM.

Turbine Stop Valve Closure, Trip Oil Pressure-Low signals are initiated by the electrohydraulic control (EHC) fluid pressure at each stop valve. Two independent pressure transmitters are associated with each stop valve. One of the two transmitters provides input to RPS trip system A; the other, to RPS trip system B. Thus, each RPS trip system receives an input from four Turbine Stop Valve Closure, Trip Oil Pressure-Low channels, each consisting of one pressure transmitter. The logic for the Turbine Stop Valve Closure, Trip Oil Pressure-Low Function is such that three or more TSVs must be closed to produce a scram.

CHANGE

This Function must be enabled at THERMAL POWER ≥ 35.4% RTP, which is the Analytical Limit. This is normally accomplished automatically by ~~pressure transmitters sensing turbine first stage pressure; therefore, to consider this Function OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER ≥ 35.4% RTP The setpoint is feedwater temperature dependent as a result of the subcooling changes that affect the turbine first stage pressure/reactor power relationship.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

10. Turbine Control Valve Fast Closure, Trip Oil Pressure—Low (continued)

with each control valve, the signal from each transmitter being assigned to a separate RPS logic channel. This Function must be enabled at THERMAL POWER ≥ 35.4% RTP. This is normally accomplished automatically by ~~pressure transmitters sensing turbine first stage pressure; therefore, to consider this Function OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER ≥ 35.4% RTP.~~ The basis for the setpoint of this automatic bypass is identical to that described for the Turbine Stop Valve Closure, Trip Oil Pressure—Low Function.

CHANGE

REACTOR POWER SIGNALS DERIVED FROM THE POWER RANGE NEUTRON MONITORING SYSTEM.

The Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

Four channels of Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Function, with two channels in each trip system arranged in a one-out-of-two logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. This Function is required, consistent with the analysis assumptions, whenever THERMAL POWER is ≥ 35.4% RTP. This Function is not required when THERMAL POWER is < 35.4% RTP since the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor Fixed Neutron Flux—High Functions are adequate to maintain the necessary safety margins.

11. Reactor Mode Switch—Shutdown Position

The Reactor Mode Switch—Shutdown Position Function provides signals, via the manual scram logic channels, that are redundant to the automatic protective instrumentation channels and provide manual reactor trip capability. This Function was not specifically credited in the accident analysis, but it is retained for the RPS as required by the NRC approved licensing basis.

The reactor mode switch is a single switch with four channels, each of which inputs into one of the RPS logic channels.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.13

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods, in LCO 3.1.3, "Control Rod OPERABILITY," and SDV vent and drain valves, in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency.

SR 3.3.1.1.14

This SR ensures that scrams initiated from the Turbine Stop Valve Closure, Trip Oil Pressure - Low and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥ 35.4% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodology are incorporated into the actual setpoint. ~~Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed at THERMAL POWER ≥ 35.4% RTP to ensure that the calibration remains valid.~~

DELETE

If any bypass channel setpoint is nonconservative (i.e., the Functions are bypassed at ≥ 35.4% RTP, ~~either due to open main turbine bypass valve(s) or other reasons~~), then the affected Turbine Stop Valve, Trip Oil Pressure - Low and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 24 months is based on engineering judgment and reliability of the components.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.2.1 Control Rod Block Instrumentation

BASES

BACKGROUND

Control rods provide the primary means for control of reactivity changes. Control rod block instrumentation includes channel sensors, logic circuitry, switches, and relays that are designed to ensure that specified fuel design limits are not exceeded for postulated transients and accidents. During high power operation, the rod withdrawal limiter (RWL) provides protection for control rod withdrawal error events. During low power operations, control rod blocks from the rod pattern controller (RPC) enforce specific control rod sequences designed to mitigate the consequences of the control rod drop accident (CRDA). During shutdown conditions, control rod blocks from the Reactor Mode Switch – Shutdown Position ensure that all control rods remain inserted to prevent inadvertent criticalities.

The purpose of the RWL is to limit control rod withdrawal to preclude a MCPR Safety Limit (SL) violation. The RWL supplies a trip signal to the Rod Control and Information System (RCIS) to appropriately inhibit control rod withdrawal during power operation equal to or greater than the low power setpoint (LPSP). The RWL has two channels, either of which can initiate a control rod block when the channel output exceeds the control rod block setpoint. The rod block logic circuitry in the RCIS is arranged as two redundant and separate logic circuits. These circuits are energized when control rod movement is allowed. The output of each logic circuit is coupled to a comparator by the use of isolation devices in the rod drive control cabinet. The two logic circuit signals are compared and rod blocks are applied when either circuit trip signal is present. Control rod withdrawal is permitted only when the two signals agree. Each rod block logic circuit receives control rod position indication from a separate channel of the Rod Position Information System, each with a set of reed switches for control rod position indication. Control rod position is the primary data input for the RWL. ~~First stage turbine pressure~~ is used to determine reactor power level, with an LPSP and a high power setpoint (HPSP) used to determine allowable control rod withdrawal distances. Below the LPSP, the RWL is automatically bypassed (Ref. 1).

THE POWER RANGE NEUTRON MONITORING SYSTEM

CHANGE

(continued)

POWER RANGE NEUTRON MONITORING SYSTEM

CHANGE

BASES

BACKGROUND (continued)

The purpose of the RPC is to ensure control rod patterns during startup are such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 10% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. The RPC, in conjunction with the RCIS, will initiate control rod withdrawal and insert blocks when the actual sequence deviates beyond allowances from the specified sequence. The rod block logic circuitry is the same as that described above. The RPC also uses the ~~turbine first stage pressure~~ to determine when reactor power is above the power at which the RPC is automatically bypassed (Ref. 1).

With the reactor mode switch in the shutdown position, a control rod withdrawal block is applied to all control rods to ensure that the shutdown condition is maintained. This function prevents criticality resulting from inadvertent control rod withdrawal during MODE 3 or 4, or during MODE 5 when the reactor mode switch is required to be in the shutdown position. The reactor mode switch has two channels, with each providing inputs into a separate rod block circuit. A rod block in either circuit will provide a control rod block to all control rods.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a. Rod Withdrawal Limiter

The RWL is designed to prevent violation of the MCPR SL and the cladding 1% plastic strain fuel design limit that may result from a single control rod withdrawal error (RWE) event. The analytical methods and assumptions used in evaluating the RWE event are summarized in Reference 2. A statistical analysis of RWE events was performed to determine the MCPR response as a function of withdrawal distance and initial operating conditions. From these responses, the fuel thermal performance was determined as a function of RWL allowable control rod withdrawal distance and power level.

The RWL satisfies Criterion 3 of the NRC Policy Statement. Two channels of the RWL are available and are required to be OPERABLE to ensure that no single instrument failure can preclude a rod block from this Function. The RWL high power function channels are OPERABLE when control rod withdrawal is limited to no more than two notches. The RWL low power function channels are OPERABLE when control rod withdrawal is limited to no more than four notches.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1.1, SR 3.3.2.1.2, SR 3.3.2.1.3, and SR 3.3.2.1.4 (continued)

DERIVED FROM THE POWER RANGE NEUTRON MONITORING SYSTEM

CHANGE (2 PLACES)

control rod block occurs. Proper operation of the RWL is verified by SR 3.3.2.1.1 which verifies proper operation of the two-notch withdrawal limit and SR 3.3.2.1.2 which verifies proper operation of the four-notch withdrawal limit. Proper operation of the RPC is verified by SR 3.3.2.1.3 and SR 3.3.2.1.4. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. As noted, the SRs are not required to be performed until 1 hour after specified conditions are met (e.g., after any control rod is withdrawn in MODE 2). This allows entry into the appropriate conditions needed to perform the required SRs. The Frequencies are based on reliability analysis (Ref. 7).

SR 3.3.2.1.5

The LPSP is the point at which the RPCS makes the transition between the function of the RPC and the RWL. This transition point is automatically varied as a function of power. This power level is ~~inferred from the first stage turbine pressure~~ (one channel to each trip system). These power setpoints must be verified periodically to be within the Allowable Values. If any LPSP is nonconservative, then the affected Functions are considered inoperable. Since this channel has both upper and lower required limits, it is not allowed to be placed in a condition to enable either the RPC or RWL Function. Because main turbine bypass steam flow can affect the LPSP nonconservatively for the RWL, the RWL is considered inoperable with any main turbine bypass valves open. The Frequency of 92 days is based on the setpoint methodology utilized for these channels.

SR 3.3.2.1.6

This SR ensures the high power function of the RWL is not bypassed when power is above the HPSP. The analytical limit for the HPSP is 70%. The power level is ~~inferred from turbine first stage pressure signals~~. Periodic testing of the HPSP channels is required to verify the setpoint to be less than or equal to the limit. Adequate margins in accordance with setpoint methodologies are included. If the HPSP is nonconservative, then the RWL is considered inoperable. Alternatively, the HPSP can be placed in the conservative condition (nonbypass). If placed

(continued)

BASES

BACKGROUND (continued)	system trips one of the two EOC-RPT breakers for each recirculation pump and the second trip system trips the other EOC-RPT breaker for each recirculation pump.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The TSV Closure, Trip Oil Pressure—Low and the TCV Fast Closure, Trip Oil Pressure—Low Functions are designed to trip the recirculation pumps from fast speed operation in the event of a turbine trip or generator load rejection to mitigate the neutron flux, heat flux, and pressure transients, and to increase the margin to the MCPR SL. The analytical methods and assumptions used in evaluating the turbine trip and generator load rejection, as well as other safety analyses that assume EOC-RPT, are summarized in References 2, 3, and 4.

To mitigate pressurization transient effects, the EOC-RPT must trip the recirculation pumps from fast speed operation after initiation of initial closure movement of either the TSVs or the TCVs. The combined effects of this trip and a scram reduce fuel bundle power more rapidly than does a scram alone, resulting in an increased margin to the MCPR SL. Alternatively, MCPR limits for an inoperable EOC-RPT as specified in the COLR are sufficient to mitigate pressurization transient effects. The EOC-RPT function is automatically disabled when ~~turbine first stage pressure is~~ THE POWER RANGE NEUTRON MONITORING SYSTEM INDICATES < 35.4% RTP.

CHANGE

EOC-RPT instrumentation satisfies Criterion 3 of the NRC Policy Statement.

The OPERABILITY of the EOC-RPT is dependent on the OPERABILITY of the individual instrumentation channel Functions. Each Function must have a required number of OPERABLE channels in each trip system, with their setpoints within the specified Allowable Value of SR 3.3.4.1.3. The actual setpoint is calibrated consistent with applicable setpoint methodology assumptions. Channel OPERABILITY also includes the associated EOC-RPT breakers. Each channel (including the associated EOC-RPT breakers) must also respond within its assumed response time.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

transmitter associated with each stop valve, and the signal from each transmitter is assigned to a separate trip channel. The logic for the TSV Closure, Trip Oil Pressure—Low Function is such that two or more TSVs must be closed to produce an EOC-PT. This Function must be enabled at THERMAL POWER ≥ 35.4% RTP. This is normally accomplished automatically by ~~pressure transmitters sensing turbine~~

CHANGE

REACTOR POWER SIGNALS DERIVED FROM THE POWER RANGE NEUTRON MONITORING SYSTEM.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Turbine Stop Valve Closure, Trip Oil Pressure - Low (continued)

~~first stage pressure; therefore to consider this Function OPERABLE, the turbine bypass valves must remain shut at ≥ 35.4% RTP.~~ Four channels of TSV Closure, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TSV Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TSV closure.

DELETE (SEE PREVIOUS PAGE)

This protection is required, consistent with the safety analysis assumptions, whenever THERMAL POWER is ≥ 35.4% RTP with any recirculating pump in fast speed. Below 35.4% RTP or with the recirculation in slow speed, the Reactor Vessel Steam Dome Pressure—High and the Average Power Range Monitor (APRM) Fixed Neutron Flux—High Functions of the Reactor Protection System (RPS) are adequate to maintain the necessary safety margins.

~~The automatic enable setpoint is feedwater temperature dependent as a result of the subcooling changes that affect the turbine first stage pressure/reactor power relationship.~~

DELETE

TCV Fast Closure, Trip Oil Pressure - Low

Fast closure of the TCVs during a generator load rejection results in the loss of a heat sink that produces reactor pressure, neutron flux, and heat flux transients that must be limited. Therefore, an RPT is initiated on TCV Fast Closure, Trip Oil Pressure—Low in anticipation of the transients that would result from the closure of these valves. The EOC-RPT decreases reactor power and aids the reactor scram in ensuring that the MCPR SL is not exceeded during the worst case transient.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

TCV Fast Closure, Trip Oil Pressure—Low (continued)

CHANGE

REACTOR POWER SIGNALS DERIVED FROM THE POWER RANGE NEUTRON MONITORING SYSTEM.

Fast closure of the TCVs is determined by measuring the EHC fluid pressure at each control valve. There is one pressure transmitter associated with each control valve, and the signal from each transmitter is assigned to a separate trip channel. The logic for the TCV Fast Closure, Trip Oil Pressure—Low Function is such that two or more TCVs must be closed (pressure transmitter trips) to produce an EOC-RPT. This Function must be enabled at THERMAL POWER ≥ 35.4% RTP. This is normally accomplished automatically by ~~pressure transmitters sensing turbine first stage pressure; therefore to consider this Function OPERABLE, the turbine bypass valves must remain shut at ≥ 35.4% RTP.~~ Four channels of TCV Fast Closure, Trip Oil Pressure—Low, with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure will preclude an EOC-RPT from this Function on a valid signal. The TCV Fast Closure, Trip Oil Pressure—Low Allowable Value is selected high enough to detect imminent TCV fast closure.

DELETE

This protection is required consistent with the analysis, whenever the THERMAL POWER is ≥ 35.4% RTP with any recirculating pump in fast speed. Below 35.4% RTP or with recirculation pumps in slow speed, the Reactor Vessel Steam Dome Pressure—High and the APRM Fixed Neutron Flux—High Functions of the RPS are adequate to maintain the necessary safety margins. ~~The turbine first stage pressure/reactor power relationship for the setpoint of the automatic enable is identical to that described for TSV closure.~~

ACTIONS

A Note has been provided to modify the ACTIONS related to EOC-RPT instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable EOC-RPT instrumentation channels provide appropriate compensatory measures for separate inoperable

(continued)

Attachment 5 to GNRO-2018/00012

GEH Nuclear Energy Report 004N6431, Revision 1, "Technical Justification of the Grand Gulf Nuclear Station Modification to Operational Bypass Signal, Replacing Turbine First Stage Pressure with APRM Neutron Flux" [Non-Proprietary]

Requests for Supplemental Information," GNRO-2013/00100, December 30, 2013 (ADAMS Accession Number ML13364A286).

12. Letter, Alan Wang (NRC) to Vice President, Operations (Entergy), "Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment Regarding Maximum Extended Load Line Limit Analysis Plus (TAC No. MF2798)," August 31, 2015 (ADAMS Accession Number ML15229A219).

13. IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," 1971.

14. NRC, "NRC Draft Regulatory Issue Summary 2017-XX Supplement to RIS 2002-22," June 27, 2017 (ADAMS Accession Number ML17102B507).

HITACHI

GE Hitachi Nuclear Energy

004N6431-NP
Revision 1
January 2018

Non-Proprietary Information – Class I (Public)

Technical Justification of the Grand Gulf Nuclear Station Modification to Operational Bypass Signal, Replacing Turbine First Stage Pressure with APRM Neutron Flux

INFORMATION NOTICE

This is a non-proprietary version of the document 004N6431-P Revision 1, which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

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REVISION SUMMARY

Revision	Revision Description
0	Initial Issue
1	Issued with revised proprietary marking suitable for submission to the NRC

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ACRONYMS

Term	Definition
ABWR	Advanced Boiling Water Reactor
AOO	Anticipated Operational Occurrence
APRM	Average Power Range Monitor
BPWS	Bank Position Withdrawal Sequence
BTP	Branch Technical Position
BWR	Boiling Water Reactor
CCF	Common Cause Failure
D3	Diversity and Defense-in-Depth
EOC	End-of-Cycle
ESBWR	Economic Simplified Boiling Water Reactor
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation Systems
FRN	Federal Register Notice
GE	General Electric
GEH	GE Hitachi Nuclear Energy
GGNS	Grand Gulf Nuclear Station
HPSP	High Power Setpoint
IEEE	Institute of Electrical and Electronic Engineers
INPO	Institute of Nuclear Power Operations
IVV	Independent Verification and Validation
LAR	License Amendment Request
LBDCR	License Basis Document Change Request
LPAP	Low Power Alarm Point
LPSP	Low Power Setpoint
LTR	Licensing Topical Report
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
NF	Neutron Flux
NRC	Nuclear Regulatory Commission
NUMAC	Nuclear Measurement, Analysis, and Control

Term	Definition
OPRM	Oscillation Power Range Monitor
PRNM	Power Range Neutron Monitor
PRNMS	Power Range Neutron Monitor System
RAI	Request for Additional Information
RIS	Regulatory Issue Summary
RPCS	Rod Pattern Control System
RPS	Reactor Protection System
RSI	Request for Supplemental Information
RTS	Reactor Trip System
RWL	Rod Withdrawal Limiter
Std	Standard
TCVFC	Turbine Control Valve Fast Closure
TFSP	Turbine First Stage Pressure
TRM	Technical Requirements Manual
TSVC	Turbine Stop Valve Closure
UFSAR	Updated Final Safety Analysis Report

1. INTRODUCTION

1.1 BACKGROUND

Entergy notified GE Hitachi Nuclear Energy (GEH) that they are preparing a License Amendment Request (LAR) to the Nuclear Regulatory Commission (NRC), seeking approval for a plant modification at Grand Gulf Nuclear Station (GGNS). The modification involves replacement of Turbine First Stage Pressure (TFSP) with Neutron Flux (NF) as the input signal for Reactor Protection System (RPS) for P-bypass and End-of-Cycle (EOC) recirculation runback, and as an input for the Rod Pattern Control System (RPCS).

The P-bypass is an operational bypass of the scram signals based on Turbine Control Valve Fast Closure (TCVFC) and on Turbine Stop Valve Closure (TSVC). These two scram signals are automatically bypassed when reactor power is below a predetermined P-bypass setting, and automatically taken out of bypass when power is above the setting.

Similarly, the signal to EOC recirculation runback causes an operational bypass, at EOC, when power is below a setting.

The RPCS enforces the Bank Position Withdrawal Sequence (BPWS) when reactor power is below the Low Power Setpoint (LPSP), and enforces the Rod Withdrawal Limiter (RWL) when reactor power is above the LPSP. Another way to state this is an operational bypass occurs for BPWS when power is above the LPSP, and for RWL when power is below LPSP. An alarm occurs when power is below the Low Power Alarm Point (LPAP) to alert operators of an impending change between BPWS and RWL. Also, when reactor power is above the High Power Setpoint (HPSP), the RWL enforces more stringent limits to rod movement.

In the original design, the detected value of TFSP was used to infer reactor power for the aforementioned functions. At GGNS, however, the TFSP exhibited recurring reliability issues in the sensing lines, plumbing, and transmitters. Between 1995 and 2014, there have been numerous times that operations were disrupted by a need for repairs: there were four power reductions, two start-ups that were halted, three shut downs, and one unnecessary scram. At other times, operations were not disrupted but there were unsuccessful efforts to repair the problem permanently during outages.

Due to these recurring negative experiences with TFSP at GGNS, Entergy sought an alternative input for the functions and identified Average Power Range Monitor (APRM) NF as a suitable candidate because it provides a direct input of reactor power. GGNS installed a Nuclear Measurement, Analysis, and Control (NUMAC) Power Range Neutron Monitor System (PRNMS) in 2012, and the NUMAC APRMs, which are part of the PRNMS, have proven to be very reliable at GGNS and multiple other sites.

It has been noted that the approved Licensing Topical Report (LTR) containing the GEH setpoint methodology (Reference 1) states that TFSP was selected for the input to P-bypass because TFSP is diverse from the systems that provide scrams based on dome pressure and NF. However, in the particular case of GGNS, no credit is taken for diversity from the NF scram.

1.2 OVERVIEW OF MODIFICATION

The modification simply involves using the existing average NF signal from the NUMAC APRM channels for an additional purpose. Each of the four APRM channels already provides, via isolators, an NF signal to the recorders in the control room. Following the modification, the same physical signal is also used as the input to P-bypass, to RPCS (HPSP, LPSP, and LPAP), and to EOC recirculation runback.

Note that in the remainder of this document, the signal will be referred to as the "input signal for the bypass."

There are no proposed changes to the hardware or software in the NUMAC APRM channels, which are already approved and installed. There are no proposed changes to the control room indications that occur when the bypasses are active. The modification does not involve changes to the systems that use the inputs (RPS, RPCS, and EOC recirculation runback) or to Technical Specifications at GGNS.

1.3 PURPOSE AND SCOPE

This document supports the GGNS LAR by providing technical justification for replacing the TFSP signal with the NF signal.

1.4 ORGANIZATION OF DOCUMENT

Sections 2 through 5 contain the main content of this document.

Section 2 provides the technical merit for using NF in lieu of TFSP as the input signal for the bypass.

Section 3 provides commentary on the statement about diversity in the GEH setpoint methodology LTR (Reference 1), which was originally written in 1986.

Section 4 explains why the modification is acceptable when evaluated using the up-to-date criteria for Diversity and Defense-in-Depth (D3). Namely, Section 4 explains why the proposed modification does not invalidate the conclusions in previously submitted D3 reports that evaluated GGNS using the criteria in Branch Technical Position (BTP) 7-19 (Reference 2).

Section 5 provides information leading to the conclusion that the modification does not result in more than a minimal increase in the likelihood of malfunction of the signal to the bypass.

2. TECHNICAL MERITS OF APRM NEUTRON FLUX SIGNAL

Using NF as an indication of reactor power has some noteworthy advantages when compared to using TFSP for the same purpose.

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The advantages of using NF, as discussed in this section of this report, make it an excellent alternative to TFSP. [[

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3. HISTORICAL STATEMENT ABOUT DIVERSITY

The following statement appears in Reference 1, the approved LTR containing the GEH instrument setpoint methodology.

> "Turbine first stage pressure has been historically used as the parameter to approximate reactor power and effect the actual trip bypass. The [RPS] design purposely chooses this parameter, as opposed to the more direct measurement of power such as neutron flux, in order to assure diversity between the TSVC and TCVFC scram functions and the neutron flux scram function."

This statement is not completely applicable as it pertains to GGNS because the GGNS Updated Final Safety Analysis Report (UFSAR) (Reference 5) does not credit the NF scram for diversity from the TSVC and TCVFC scrams. Rather, it credits the reactor vessel high-pressure trip signal.

The UFSAR (Reference 5) (Section 7.2.1.1.4.2, paragraph d) states the following about TSVC (emphasis added).

> "Diversity of trip initiation for increases in reactor vessel pressure due to termination of steam flow by turbine stop valve or control valve closure is provided by <u>reactor vessel high-pressure trip signals</u>. A closure of the turbine stop valves or control valves at steady-state conditions would result in an increase in reactor vessel pressure. If a scram was not initiated from these closures, a scram would occur from high reactor vessel pressure. <u>Reactor vessel high pressure</u> is an independent variable and for this condition provides diverse trip initiating circuits for the protective action (scram)."

In the discussion about the TCVFC, the UFSAR (Reference 5) (Section 7.2.1.1.4.2, paragraph e) simply points back to the discussion just quoted.

> "The discussion of diversity for turbine control valve fast closure is the same as that for turbine stop valve closure provided in subsection 7.2.1.1.4.5 and paragraph d. above."

Also, when the UFSAR (Reference 5) discusses redundancy and diversity of RPS scram inputs (Section 7.2.1.1.4.5), it identifies TSVC and TCVFC scrams as anticipatory of reactor vessel high-pressure scram, but not anticipatory of NF scram (emphasis added).

> "[The] main steam line isolation valve closure, turbine stop valve closure, and turbine control valve fast closure are <u>anticipatory of a reactor vessel high-pressure</u> and are separate inputs to the system."

Based on the above statements from the UFSAR (Reference 5), the historical and general statement found in the GEH setpoint methodology LTR is not completely applicable for GGNS. At GGNS, the NF scram is not credited for diversity, and therefore the proposed modification does not result in a reduction of diversity.

4. BRANCH TECHNICAL POSITION 7-19

The NUMAC Power Range Neutron Monitor (PRNM) LTR (Reference 6) addressed diversity by generically identifying diverse trip functions for UFSAR events, and requiring licensees to make a statement of applicability. During NRC review of the GGNS LAR for NUMAC PRNMS (Reference 7), the NRC staff issued a series of Requests for Additional Information (RAIs) concerning D3. The scope of some of the RAIs involved demonstrating that the modification met the acceptance criteria in NRC BTP 7-19. The conclusion in the licensee response (Reference 8) is that GGNS, with NUMAC PRNMS installed, meets the acceptance criteria. Additionally, in the NRC safety evaluation (Reference 9), the NRC staff determined that the proposed change provides sufficient D3 to satisfy the acceptance criteria.

Subsequently, GGNS submitted an LAR for Maximum Extended Load Line Limit Analysis Plus (MELLLA+) (Reference 10). As part of that review process, the NRC staff issued Requests for Supplemental Information (RSIs), including an RSI about D3. In the licensee response (Reference 11), content from the previously submitted RAI response (Reference 8) was revisited in light of MELLLA+ operations. It was found that MELLLA+ did not alter the conclusions about D3, and the MELLLA+ LAR was later approved (Reference 12).

This section revisits the D3 content from those submittals in light of the current proposed modification to replace TFSP with APRM NF as the input signal for the bypass. This section explains why replacing TFSP with NF does not alter the conclusion that the proposed change provides sufficient D3 to satisfy the acceptance criteria in BTP 7-19.

4.1 POSTULATED CCF FOR CURRENT PROPOSED MODIFICATION

In Reference 8, which included the D3 evaluation for the entire PRNMS modification, the postulated Common Cause Failure (CCF) was one that completely impairs all functionality of the PRNMS yet the failure provided no advanced notice of trouble prior to a transient. During a transient, the postulated CCF caused the system to fail to provide the correct instrument responses from all four channels and also provide potentially misleading information. In Reference 11, when D3 was revisited for MELLLA+, the postulated CCF was the same.

For evaluation of the current modification, however, the above postulated CCF is not appropriate. The current modification only involves an analog signal, provided by PRNMS, that is proportional to NF, and is used by existing hardware (not PRNMS) to apply or remove the operational bypass. Also, [[

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4.2 DETECTABILITY OF POSTULATED CCF

Considering the operational experience with analog signals in general and NUMAC APRM in particular, the two scenarios that arise from these two postulated CCFs are not plausible. Additionally, considering the procedural checks that are in place to ensure the APRMs are indicating power correctly, the notion that these scenarios could occur and also go undetected is completely unrealistic.

During power ascension, GGNS Technical Specifications (Reference 3) require calibrating the APRM after the thermal power is at least 21.8% of rated thermal power for 12 hours. Additionally, GGNS informed GEH that their procedures require calibration of the APRMs at 10% - 12% reactor power and again at approximately 18% reactor power. The procedures also require verification of the calibration between 22% and 24% power. [[
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The GGNS Technical Specifications (Reference 3) and TRM (Reference 4) also require calibrating the APRM weekly after power is at least 21.8% of rated thermal power. This alone makes it essentially impossible for CCF #2 to occur and remain undetected. In addition, there would be spurious indications that make the failure detected immediately. The P-bypass is annunciated in the control room, as required by Institute of Electrical and Electronic Engineers (IEEE) Standard (Std) 279-1971 (Reference 13). If this bypass was correctly removed during power ascension but incorrectly applied later due to [[
]] the bypass would be annunciated. The same would be true of the low-pressure alarm, which is also annunciated. Moreover, depending on the nature of the failure, there could be an APRM downscale, which causes an indication. [[
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Based on these considerations, there is a high degree of certainty the postulated APRM CCFs will be detected and addressed, eliminating them as items of concern. As stated in BTP 7-19 (Reference 2), "the primary concern (with CCF in digital systems) is that an undetected failure within a digital safety system could prevent proper system operation. A failure or fault that is detected can be addressed; however, failures that are non-detectable may prevent a system actuation that is necessary. Consequently, non-detectable faults are of concern."

For the proposed modification, such a CCF that remains undetected is unrealistic. Therefore, no action is necessary because there is no vulnerability to the consequences of the CCF remaining undetected.

4.3 DISCUSSION OF THE NINE ACCEPTANCE CRITERIA IN BTP 7-19

Table 1 below revisits each of the nine criteria in BTP 7-19, similar to what was done in Reference 11 during the MELLLA+ application, and provides an evaluation as it relates to the current proposed modification.

Table 1 – D3 Acceptance Criteria

BTP 7-19 Criterion	Evaluation of Current Modification
(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.*	This criterion requires an evaluation of each Anticipated Operational Occurrence (AOO) concurrent with the postulated CCF. Table 8-1 of Reference 8 provided such an evaluation for each AOO concurrent with the CCF that totally impaired PRNMS. The Reference 11 table revisited that content for MELLLA+ and also discussed thermal hydraulic instability in additional detail. The current modification does not alter any conclusions about AOOs because, as discussed in Section 4.2, a postulated APRM CCF that could affect the bypass input in a way that causes a safety function to be defeated will be detected and addressed. The analysis assumes proper operation of the operational bypass for the following AOOs: 15.2.2 Generator Load Rejection 15.2.3 Turbine Trip 15.4.2 Rod Withdrawal at Power The bypass has no effect on the Oscillation Power Range Monitor (OPRM) or plant response to thermal hydraulic instability. Acceptance Criterion (1) is satisfied.

BTP 7-19 Criterion	Evaluation of Current Modification
(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.*	This criterion requires an evaluation of each accident concurrent with the postulated CCF. Table 8-1 of Reference 8 provided such an evaluation for each accident concurrent with the CCF that totally impaired PRNMS. The Reference 11 table revisited that content for MELLLA+. The current modification does not alter the conclusions about accidents because, as discussed in Section 4.2 of this report, a postulated APRM CCF that could affect the bypass input in a way that causes a safety function to be defeated will be detected and addressed. The analysis assumes proper operation of the operational bypass for the one accident: 15.4.9 Control Rod Drop Accident Acceptance Criterion (2) is satisfied.
(3) When a failure of a common element or signal source shared by the control system and reactor trip system (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 current modification does not alter the conclusion from the previous submittals: PRNMS 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 a reactor trip. Therefore, the type of CCF described in this criterion cannot occur in the upgrade system. Acceptance Criterion (3) is satisfied.

BTP 7-19 Criterion	Evaluation of Current Modification
(4) When a failure of a common element or signal source shared by the control system and Engineered Safety Features Actuation Systems (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 current modification does not alter the conclusion from the previous submittals: PRNMS 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.
(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 current modification does not alter the conclusion from the previous submittals: PRNMS does not rely on or receive any input from the monitoring and display echelon; therefore, a failure in the monitoring and display systems will not propagate to the PRNMS. If the failure in the monitoring and display system results in an operator-induced transient, the automatic protective functions of the PRNMS are available for compensation. Acceptance Criterion (5) is satisfied.

BTP 7-19 Criterion	Evaluation of Current Modification
(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 RTS and ESFAS functions. The current modification does not alter the conclusion from the previous submittals: This criterion is not applicable to the PRNMS upgrade. The evaluation performed for Acceptance Criteria (1) and (2) demonstrates that if a CCF occurs in the PRNMS, the plant is able to cope without relying on a manual scram or 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 PRNMS and not affected by the proposed modification in any way. Acceptance Criterion (6) is not applicable to the modification.
(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).	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 current modification does not alter the conclusion from the previous submittals: The NUMAC platform is not present in any part of the RTS except the PRNMS and is not present in the ESFAS. These designs are not affected by the proposed modification, and these systems are not vulnerable to the postulated CCF in the PRNMS. Therefore, Acceptance Criterion (7) is not applicable to the modification.

BTP 7-19 Criterion	Evaluation of Current Modification
(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.*	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 the RTS except the PRNMS, and is not present in the ESFAS. These designs are not affected by the proposed modification, and these systems are not vulnerable to the postulated CCF in the PRNMS. Acceptance Criterion (8) is not applicable to the modification.
(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 the RTS except the PRNMS, and is not present in the ESFAS. These designs are not affected by the proposed modification, and these systems are not vulnerable to the postulated CCF in PRNMS. Acceptance Criterion (9) is not applicable to the modification.

5. LIKELIHOOD OF A MALFUNCTION

This section provides information supporting the conclusion that the modification does not result in more than a minimal increase in the likelihood of a malfunction of input signal for the bypass. Based on the guidance in (draft) Regulatory Issue Summary (RIS) 2017-xx (Reference 14), this section qualitatively assesses the design attributes of the NUMAC APRM, the quality design process GEH used to develop the NUMAC APRM, and operating experience of NUMAC APRM.

It is acknowledged that Reference 14 is not issued. However, there was an NRC public meeting about it on October 25, 2017, following the Federal Register Notice (FRN) public comment period. The NRC staff provided general updates on the public resolution and expanded on items with "greater impact." The indication was that resolving public comments would not lead to major changes to the document, such as adding to or deleting from the three broad categories in a qualitative assessment. Considering this, and that something similar does not already exist, the draft guidance is appropriate to use for the present purposes.

5.1 DESIGN ATTRIBUTES

The approved LTR and supplement for the safety-related NUMAC PRNMS (Reference 6), which includes the safety-related APRMs, demonstrated that the PRNMS design was acceptable based on applicable industry standards and regulatory guides at the time.

During the GGNS PRNMS LAR (Reference 7) review, which occurred from 2010 to 2012, the NRC staff provided RAIs about the design. The responses to these RAIs provided updated or new information about the PRNMS, demonstrating that its design meets current applicable standards and regulatory guides, including those that came into force after the LTR approval. The NRC safety evaluation (Reference 9) concluded that the LTR and the responses to the RAIs, collectively, addressed the up-to-date criteria for topics such as qualification, redundancy, electrical separation, communication independence, and reliability.

The proposed plant modification does not involve any changes to the PRNMS design. Therefore, concerning these important design attributes, the modification does not call into question the conclusions already documented in the NRC safety evaluation.

5.2 QUALITY DESIGN PROCESS

The PRNMS LTR (Reference 6) prescribed the software development process for the system, and the NRC approval of the LTR addressed that development process. During the review of the LAR to install PRNMS at GGNS (Reference 7), the NRC staff provided several RAIs about the GEH process for software development and testing. RAIs addressed topics such as the independence of the Independent Verification and Validation (IVV) organization, the rigor of tool evaluation, and configuration control.

In the responses to those RAIs, which involved evaluating the initial GEH development process against more recent standards, a gap was identified. It was found that the level of independence in GEH software module testing and integration testing did not meet the current standards. As a result, GEH performed compensatory measures to fill the gap. In Section 3.7 of the safety evaluation (Reference 9), the NRC approved the development process, as supplemented by the compensatory measures, for the GGNS NUMAC PRNMS.

The proposed plant modification does not involve any changes to the PRNMS software. Therefore, concerning the software and the development process, the modification does not call into question the conclusions already documented in the NRC safety evaluation (Reference 9).

5.3 OPERATING EXPERIENCE

The GGNS system has been in service for over five years since its installation in 2012. Moreover, the NUMAC PRNMS is installed in 18 plants (including GGNS) in the United States, in addition to 12 plants in overseas locations. In total, the NUMAC PRNMS has over 400 years of operating experience.

GEH is not aware of any occasions when the NF signal from the NUMAC PRNMS had a malfunction. Although the signal is not used extensively as an input to the bypass, the same signal is used extensively as an input to meters and recorders. The GEH staff who field questions from their technical counterparts at operating plants do not recall any instances when a question involved the NF (or simulated thermal power) output signals to the meters or recorders. Also, no reports about this signal were found during a search of the operating experience database maintained by Institute of Nuclear Power Operations (INPO).

5.4 ASSERTIONS

Based on the design attributes, development process, and operating experience of NUMAC PRNMS, it is a reliable system. The current plant modification, which results in using NF from the NUMAC APRMs (part of PRNMS) as the signal for the bypass, does not involve any changes to PRNMS. Therefore, the NUMAC APRMs are a reliable source of the signal for the bypass.

Moreover, as discussed in Section 1.1, the TFSP signal at GGNS was a source of recurring problems. Therefore, using NF is the more reliable signal.

6. CONCLUSIONS

The following conclusions support the acceptability of the proposed modification:

- The NF has several important technical advantages over TFSP as the input signal for the bypass. It is the current thinking at GEH that using a different signal, simply because NF is also a scram input, does not outweigh the technical advantages. Moreover, in the particular case of GGNS, the UFSAR does not even credit the flux scram for diversity from the TCFVC and TSVC scrams.
- The proposed use of NF does not cause any violations of the acceptance criteria in BTP 7-19 (Reference 2).
- The system that provides NF, the NUMAC PRNMS, can be relied upon to perform this function. The system has design attributes that make it reliable, it was developed using an approved quality process, and has extensive operating experience. In the case of GGNS, the NF signal is more reliable than TFSP.

7. REFERENCES

1. GE Hitachi Nuclear Energy, "General Electric Instrument Setpoint Methodology," NEDC-31336P-A, September 1996.
2. NUREG-0800, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems," Branch Technical Position 7-19, Revision 6, July 2012 (ADAMS Accession Number ML110550791).
3. GGNS Technical Specifications, Amendment Number 191, July 18, 2012.
4. GGNS TRM, License Basis Document Change Request (LBDCR) 12035, July 19, 2012.
5. GGNS UFSAR, Revison 2016-00.
6. NUMAC PRNM LTRs and Approvals:
   a. GE Nuclear Energy, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," NEDC-32410P-A Volume 1, October 1995.
   b. GE Nuclear Energy, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," NEDC-32410P-A Volume 2, October 1995.
   c. GE Nuclear Energy, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC PRNM) Retrofit Plus Option III Stability Trip Function," NEDC-32410P-A Supplement 1, November 1997.
7. Letter, Michael A. Krupa (Entergy) to NRC Document Control Desk, "License Amendment Request Power Range Neutron Monitoring System Upgrade," GNRO-2009-00054, November 3, 2009 (ADAMS Accession Number ML093140463).
8. Letter, Michael A. Krupa (Entergy) to NRC Document Control Desk, "Responses to NRC Requests for Additional Information Pertaining to License Amendment Request for Power Range Neutron Monitoring System (TAC No. ME2531)," GNRO-2011/00039, May 26, 2011 (ADAMS Accession Number ML111460590).
9. Letter, Alan Wang (NRC) to Vice President, Operations (Entergy), "Grand Gulf Nuclear Station, Unit 1 - Issuance of Amendment RE: Power Range Neutron Montoring System Replacement (TAC No. ME2531)," March 28, 2012 (ADAMS Accession Number ML120400319).
10. Letter, Bryan S. Ford (Entergy) to NRC Document Control Desk, "Maximum Extended Load Line Limit Analysis Plus (MELLLA+) License Amendment Request," GNRO-2013/00012, September 25, 2013 (ADAMS Accession Number ML13269A140).
11. Letter, Kevin Mulligan (Entergy) to NRC Document Control Desk, "Maximum Extended Load Line Limit Analysis Plus (MELLLA+) License Amendment Request - Responses to

Requests for Supplemental Information," GNRO-2013/00100, December 30, 2013 (ADAMS Accession Number ML13364A286).

12. Letter, Alan Wang (NRC) to Vice President, Operations (Entergy), "Grand Gulf Nuclear Station, Unit 1 – Issuance of Amendment Regarding Maximum Extended Load Line Limit Analysis Plus (TAC No. MF2798)," August 31, 2015 (ADAMS Accession Number ML15229A219).

13. IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations," 1971.

14. NRC, "NRC Draft Regulatory Issue Summary 2017-XX Supplement to RIS 2002-22," June 27, 2017 (ADAMS Accession Number ML17102B507).

Attachment 6 to GNRO-2018/00012

Affidavit

GE-Hitachi Nuclear Energy Americas LLC

AFFIDAVIT

I, Lisa K. Schichlein, state as follows:

(1) I am a Senior Project Manager, NPP/Services Licensing, Regulatory Affairs, GE-Hitachi Nuclear Energy Americas LLC (GEH), and 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 GEH proprietary report 004N6431-P, "Technical Justification of the Grand Gulf Nuclear Station Modification to Operational Bypass Signal, Replacing Turbine First Stage Pressure with APRM Neutron Flux," Revision 1, dated January 2018. GEH proprietary information in 004N6431-P Revision 1 is identified by a dotted underline inside double square brackets. [[This sentence is an example.{3}]] GEH proprietary information in large objects is identified by double square brackets before and after the object. In each case, the superscript notation {3} refers to Paragraph (3) of this affidavit, which 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 U.S.C. §552(b)(4), and the *Trade Secrets Act*, 18 U.S.C. §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 F.2d 871 (D.C. Cir. 1992), and Public Citizen Health Research Group v. FDA, 704 F.2d 1280 (D.C. 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 a license from GEH constitutes a competitive economic advantage over other companies;

b. Information that, if used by a competitor, would reduce its expenditure of resources or improve its 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, resulting in potential products to GEH;

d. Information that discloses trade secret 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 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 for proprietary or confidentiality agreements or both 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.

(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) is classified as proprietary because it contains details of GEH's instrument setpoint methodology for the GEH Boiling Water Reactor (BWR). The development of this methodology, along with the testing, development, and approval, was achieved at a significant cost to GEH.

The development of the design and licensing methodology along with the interpretation and application of the analytical results is derived from the extensive experience databases that constitute 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 is true and correct.

Executed on this 29th day of January 2018.

Lisa K. Schichlein

Lisa K. Schichlein
Senior Project Manager, NPP/Services Licensing
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