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August 11, 1993

U. S. Nuclear Regulatory Commission
Mail Station P1-37
Washington D. C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
Relocation of Accident Monitoring Instrumentation
Proposed Change to the Operating License (PCOL-93/09)

Reference: NUREG-1434, Revision 0 of the Improved Standard Technical Specifications

GNRO-93/00097

Pursuant to 10CFR50.90, Entergy Operations, Inc. is hereby proposing to amend Operating License NPF-29 for the Grand Gulf Nuclear Station (GGNS) by incorporating the enclosed changes into the plant Technical Specifications. Specifically, the proposed changes would amend Technical Specification Section 3/4.3.7.5, entitled "Accident Monitoring Instrumentation" by relocating certain accident monitoring instrumentation from the Technical Specifications to our administrative control.

Guidance from the Improved Standard Technical Specifications (NUREG-1434, Revision 0) recommends that accident monitoring instrumentation not designated as Regulatory Guide (R.G.) 1.97 Category I or Type A and their associated LCO and surveillances may be relocated from the Technical Specifications to other controlled plant documents. Consistent with this guidance, we propose to relocate accident monitoring instrumentation currently in the GGNS Technical Specification that is not designated as R. G. 1.97 Category I or Type A. Accident monitoring instrumentation that is designated as R.G. 1.97 Category I or Type A is being retained in the Technical Specifications.

This proposed amendment is being submitted as part of the cost beneficial licensing action (CBLA) program established within NRR where increased priority is granted to licensee requests for changes requiring staff review that involve high cost without a commensurate safety benefit. Ultimately, our goal is to evaluate the accident monitoring instrumentation against regulatory and safety requirements and eliminate overly restrictive redundancy under 10CFR50.59. Through reduced maintenance and surveillance activities, GGNS expects accrued cost reductions on the order of $900,000.00 over the remaining operating life of the plant resulting from elimination of redundant monitoring capability while maintaining an equivalent level of protection.

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Since the proposed changes are consistent with the Improved Standard Technical Specifications, we believe that the NRC resources necessary to review this proposal are not significant. Nonetheless, in order to provide the NRC with a means to prioritize the Entergy Operations CBLA requests, should a resource conflict occur between this request and any other Entergy Operations CBLA proposal, the other proposal should take precedence.

Attachment 2 provides a detailed description of the proposed changes, justification for the changes, and the No Significant Hazards Consideration in accordance with 10CFR50.92. As stipulated by the provisions of 10CFR50.4, Attachment 3 contains the marked up original Technical Specification and bases pages reflecting the proposed change. A copy of the proposed change is provided for information in Attachment 4.

This proposed change has been reviewed and accepted by the Plant Safety Review Committee and the Safety Review Committee. A copy of this proposed operating license change has been provided to the State of Mississippi in compliance with the requirements of 10CFR50.91(B)(7)(b).

This amendment request is being filed without the required affirmation. The affirmation referenced in Attachment 1 as required by 10CFR50.30(b) will be provided promptly.

Should you have any questions, please contact C. E. Brooks at (601) 437-6555.

for

CRH/CEB

attachments: 1. Affirmation per 10CFR50.30 (Not included)
2. GGNS PCOL-93/09
3. Mark-up of Affected Technical Specifications Pages
4. Information Copy of Proposed Technical Specifications Pages

cc: (See Next Page)

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This amendment request is being filed without the required affirmation. The affirmation referenced in Attachment 1 will be provided promptly.

PROPOSED CHANGE TO THE OPERATING LICENSE

ACCIDENT MONITORING INSTRUMENTATION

(GGNS PCOL-93/09)

A. SUBJECT

PCOL 93/09: Relocation of Accident Monitoring Instrumentation

Affected Technical Specification Pages: 3/4 3-74, 3/4 3-76, 3/4 4-5, and 3/4 4-6

Corresponding Affected Technical Specifications: Table 3.3.7.5-1 "Accident Monitoring Instrumentation", Table 4.3.7.5-1 "Accident Monitoring Instrumentation Surveillance Requirements", Sections 3.4.2.1.b & c, Section 4.4.2.1.1, and Sections 4.4.2.1.1.a & b.

B. BACKGROUND

The purpose of the accident monitoring instrumentation is to display plant variables that provide information required by the control room operator during and following accident conditions. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis events. The instrumentation that monitor these variables is designated as Category I or Type A in accordance with the recommendations of R. G. 1.97.

As requested by Generic Letter 82-33 (NUREG-0737, Supplement 1), GGNS provided a report to the NRC describing how the accident monitoring instrumentation installed at GGNS meets the recommendations of R.G. 1.97 as applied to emergency response facilities . The GGNS response was provided in letters dated February 28, 1985 and February 14, 1986, respectively (References 3 & 4). NRC acceptance of the GGNS position is documented in the staff's SER dated January 12, 1987 (Reference 5).

C. DISCUSSION

The intent of R.G. 1.97 is to ensure that necessary and sufficient instrumentation exists in a nuclear power facility for assessing plant and environmental conditions both during and following an accident as required by 10 CFR 50, Appendix A (General Design Criteria 13, 19, and 64). R.G. 1.97 utilizes three distinct categories for stipulating requirements for accident monitoring instrumentation depending on importance to safety. Category I instruments are assigned the most stringent requirements and are used to measure key variables (i.e., variables directly indicating accomplishment of a safety function). Category 2 instruments are assigned the next most stringent requirements and generally applies to instrumentation designed to indicate system operating status. Category 3 instruments are assigned the least stringent requirements and applies to instrumentation that serves in a backup or diagnostic capacity.

Currently the GGNS TS contain a listing of accident monitoring instrumentation along with the associated operability and surveillance requirements for these instruments. Many of the instruments listed are beyond the scope of the instruments that are required to be included in TS per guidance from the Improved Standard Technical Specifications, (Reference 1). GGNS proposes to modify the TS by relocating accident monitoring instruments that are not designated as R. G. 1.97 Category I or Type A from the TS to our administrative control. Upon approval of this proposed change, our ultimate goal is to evaluate the relocated instrumentations under the provisions of 10CFR50.59 and eliminate redundant overly restrictive monitoring capability. The instrumentation proposed for relocation is not solely relied upon for performing any safety related activity.

C. **DISCUSSION (Continued)**

Accident monitoring instrumentation that GGNS is proposing to relocate from the Technical Specifications is discussed below.

The process and effluent radiological monitoring and sampling systems are described in Sections 7.5 and 11.5 of the GGNS UFSAR. This description provides the design objective and criteria for determining system designation as either 1) Instrumentation systems required for safety, or 2) Instrumentation systems required for plant operation. The radiation monitors being proposed for relocation from the technical specifications are designated as "Instrumentation Systems Required for Plant Operation". The main objective of these instrumentation systems is to provide plant operating personnel with measurement of the content of radioactive material in all effluent and important process streams. Additional discussion of this instrumentation is provided in the justification (Section D).

The Drywell/Containment differential pressure is described in the GGNS UFSAR Section 7.5. This pressure indication provides a measurement of drywell pressure referenced to the pressure inside containment. The purpose of this indication is to verify that blowdown is complete following a LOCA and that the drywell purge compressors have repressurized the drywell. Additional discussion of this instrumentation is provided in the justification (Section D).

The safety/relief valve tail pipe pressure switch position indicators are discussed in Section 7.3 of the GGNS UFSAR. These indicators provide verification that the safety/relief valves have opened. These indicators were installed per NUREG-0578 as part of the short-term lessons learned from TMI. Additional discussion of this instrumentation is provided in the justification (Section D).

Specifically GGNS proposes to make the following changes to the Technical Specifications following the guidance of NUREG-1434, Revision 0:

1. Relocate the following accident monitoring instrumentation from TS Table 3.3.7.5-1 as well as the LCO requirements to our administrative control:

   * Drywell/Containment Differential Pressure
   * Safety/Relief Valve Tail Pipe Pressure Switch Indicators
   * Containment Ventilation Exhaust Radiation Monitor
   * Off-gas and Radwaste Bldg. Ventilation Exhaust Radiation Monitor
   * Fuel Handling Area Ventilation Exhaust radiation Monitor
   * Turbine Bldg. Ventilation Exhaust Radiation Monitor
   * Standby Gas Treatment System A & B Exhaust Radiation Monitor

2. Relocate the corresponding surveillance requirements listed for these accident monitoring instruments in Table 4.3.7.5-1 to plant administrative controls.

   Relocating the controls on these surveillance requirements involves no significant changes to the surveillance and operability requirements currently contained in the GGNS Technical Specifications. The details of these surveillance requirements are currently located in plant procedures. GGNS adheres to a policy of verbatim compliance with plant procedures.

This information will be controlled by the administrative requirements specified in Technical Specifications 6.8 and 6.5.3. Those requirements include review of changes for unreviewed safety questions in accordance with the provisions of 10CFR50.59. Such changes are reported to the NRC in the annual report submitted pursuant to 10CFR50.59. These changes, therefore constitute an administrative revision only.

These changes will provide GGNS the necessary flexibility to pursue evaluation of redundancy in monitoring capability in the accident monitoring system under the provisions of 10CFR50.59.

D. JUSTIFICATION

GGNS proposes to relocate the accident monitoring instrumentation listed in Table 1 from the plant Technical Specifications to our administrative control. As denoted in Table 1, these instruments are not designated as R. G. 1.97 Category I or Type A. Relocating this instrumentation is consistent with the guidance of the Standard Improved Technical Specifications (NUREG 1434, Revision 0).

On February 6, 1987, the NRC published its Interim Policy Statement on Technical Specification Improvements for Nuclear Power Plants in the Federal Register (52 FR 3788). In late 1987, based on the Interim Policy Statement, each of the four nuclear steam supply system (NSSS) owners groups submitted proposals identifying requirements in the existing Standard Technical Specifications (STS) that could be relocated from the TS to licensee controlled documents.

The staff reviewed these submittals and published its conclusion in the report "NRC Staff Review of Nuclear Steam Supply Vendors Owners Groups' Application of the Commission's Interim Policy Statement Criteria to Standard Technical Specifications" (Split Report) dated May 9, 1988.

The NRC Interim Policy Statement provides criteria to be utilized in determining which requirements need to be governed by TS. The goal is to assure that TS requirements are consistent with 10CFR50.36 and have a sound safety basis. The split report identified which STS requirements must be retained in the new STS (having met one or more criteria) and those requirements which could be relocated (having met none of the criteria).

Following the guidance of the split report, the owners groups proposed improved STS which were subsequently approved and published by the staff as improved STS NUREG reports.

NEDO-31466 "Technical Specification Screening Criteria Application and Risk Assessment" dated November 1987, designated the accident monitoring instrumentation listed in Table 1 as instrumentation that may be relocated. This was reviewed and approved by the staff in the split report. Therefore, GGNS proposes to relocate the accident monitoring instrumentation listed in Table 1, consistent with the NRC approved TS improvements. Upon approval of this change, the accident monitoring instrumentation currently listed in the technical specifications will be relocated to our administrative control and governed under the rigorous provisions of 10CFR50.59.

Part A of Table 1 lists the following accident monitoring instrumentation for relocation from the Technical Specifications to another controlled plant document:

* Drywell/Containment Differential Pressure

As previously stated, the drywell/containment differential pressure indication verifies that LOCA blowdown is complete and that the drywell purge compressors have repressurized the drywell. Instrument redundancy is provided by utilizing two drywell/containment pressure signals that are transmitted from separate transmitters and recorded on two separate one pen recorders in the control room. The power sources are from two independent dc buses through two independent inverters. The drywell/containment differential pressure indicators are designated as R.G 1.97 Category 2, Type D variable. As such, these accident monitoring instruments meet the NUREG-1434 criteria for removal from the technical specifications to another controlled plant document.

Part B of Table 1 lists the following accident monitoring instrumentation for relocation from the technical specifications to another controlled plant document:

* Safety/Relief Valve Tail Pipe Pressure Switch Position Indicators

Safety/Relief valve position indication is provided by sensing the pressure in each SRV tailpipe via the tailpipe pressure switches. This instrumentation consists of a pressure switch connected by a sensing line connected to the discharge piping of the safety/relief valve. An open SRV pressurizes the discharge line and the sensing line to the pressure switch, actuating the switch. The output of the pressure switch actuates a relay in the control room. The relay contacts input to the annunciator in the control room, to the process computer, and to an indicator light on the control room panel indicating SRV position. The pressure switch is designed for LOCA conditions and a power supply monitor and annunciator are provided to alarm on loss of power to this system. The system is designed such that a leaking pilot valve or SRV will not actuate the system and providc misleading indications. This instrumentation is designated as R. G. 1.97 Category 2, Type D, thereby meeting the NUREG-1434 criteria for relocation to another controlled plant document.

Table 1 Part C lists the following effluent release radioactivity monitors proposed for relocation:

* Containment Ventilation Exhaust Radiation Monitor
* Off-gas and Radwaste Bldg. Ventilation Exhaust Radiation Monitor
* Fuel Handling Area Ventilation Exhaust radiation Monitor
* Turbine Bldg. Ventilatio ı Exhaust Radiation Monitor

Each of the radiation monitoring systems listed in Table 1 Part C consist of a micro-processor based normal range radioactivity monitor, a micro-processor based accident range radioactivity monitor, a single flow monitoring and isokinetic sampling (FM&IS) unit located in the exhaust duct, an isokinetic sample panel, and a redundant stack flow monitoring panel. In addition to the micro-processor based system, a data acquisition module, and a central control terminal with a report generating computer interface located in the control room.

Should the radioactivity levels exceed the normal range monitor's capability (i.e., accident conditions), a dedicated sample probe for the accident range monitor will provide monitoring capability of the gaseous effluent at the higher ranges.

The operating ranges of the monitors have sufficient overlap to permit continuity of measurement upon changing from the normal to accident ranges.

A redundant radioactivity monitoring system consisting of a GE constant volume radioactivity monitoring system is used to constantly monitor airborne radioactivity. This extends the overall system capability by providing additional indication of airborne radioactivity, or in the event of FM&IS unit failure, the redundant capability to monitor radioactivity. In the event of both FM&IS failure and failure of the GE radiation monitor, provision have been made to obtain grab samples for laboratory analysis

Part D of Table 1 lists the following effluent release radioactivity monitors for relocation from the technical specifications to another controlled plant document:

* Standby Gas Treatment A & B Exhaust Ventilation Monitors

The Standby Gas Exhaust Treatment Monitors have the same configuration as the Containment, Offgas/Radwaste, and Fuel Handling Exhaust Ventilation Monitors, with the following exceptions:

1) Standby Gas normally operates during accident conditions; therefore, the SGTS radioactivity monitoring system will operate during accident and recovery conditions. The SGTS A & and B exhaust ventilation radiation monitoring systems are each powered from a Class 1E power supply.

2) Each of the SGTS effluent radioactivity monitoring systems will be manually initiated by the operator. Initiation of the radioactivity monitoring system will automatically start the isokinetic sampling portion of the system with the exception of the vacuum pump which may be started manually or automatically on initiation of SGTS.

3) The SGTS radioactivity monitoring systems do not have an associated GE system.

4) Any portion of the SGTS effluent radioactivity monitoring system which penetrates the boundary of the SGTS is designed to the seismic criteria of the exhaust duct.

5) The SGTS radioactivity monitoring system annunciates at the data acquisition module, and at both central control terminals.

6) Grab sample points are located on plant elevation 139 feet in the auxiliary building that permit on site analysis during normal and accident conditions.

TABLE 1

ACCIDENT MONITORING INSTRUMENTATION
PROPOSED FOR RELOCATION FROM THE PLANT TECHNICAL SPECIFICATIONS
(PCOL 93-09)

	INSTRUMENT	R. G. 1.97 TYPE/CATEGORY
PART A	Drywell/Containment Differential Pressure	D/2
PART B	Safety/Relief Valve Tail Pipe Pressure Switch Position Indicators	D/2
PART C	Containment Ventilation Exhaust Radiation Monitor	E/2
	Off-gas and Radwaste Bldg. Ventilation Exhaust Radiation Monitor	E/3
	Fuel Handling Area Ventilation Exhaust Radiation Monitor	E/2
	Turbine Bldg. Ventilation Exhaust Radiation Monitor	E/3
PART D	Standby Gas Treatment System A & B Exhaust Radiation Monitors	E/3

NO SIGNIFICANT HAZARDS CONSIDERATIONS

Entergy Operations, Inc. is proposing that the GGNS Technical Specifications be amended to relocate accident monitoring instrumentation not designated as R.G. 1.97 Category I or Type A from the technical specifications to plant administrative control.

The Commission has provided standards for determining whether a no significant hazards consideration exists as stated in 10CFR50.92(C). A proposed amendment to an operating license involves a no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Entergy Operations, Inc. has evaluated this proposed Technical Specification change against the no significant hazards criteria and has determined that it involves no significant hazards consideration. This determination has ben performed in accordance with the criteria set forth in 10CFR50.92. The following evaluation is provided for the three categories of the significant hazards consideration:

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

   The proposed change relocates accident monitoring instrumentation not designated as R.G. 1.97 Category I or Type A from the Technical Specifications to plant administrative control, consistent with the NRC Interim Policy Statement on Technical Specification Improvement. Criterion 1 of the Policy Statement indicates that the Technical Specifications should include installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. This criterion is intended to ensure that the technical specifications control those instruments specifically installed to detect excessive reactor coolant system leakage. This criterion is not interpreted to include instrumentation used to detect precursors to reactor coolant pressure boundary leakage (i.e., non-category 1 or non-type A Reg. Guide. 1.97 variables). The proposed change does not affect any material condition of the plant that could directly contribute to causing or mitigating the effects of an accident.

   The applicable surveillance requirements (operability testing, channel checks, and calibration requirements) proposed for relocation will be adequately controlled via the administrative requirements of Technical Specification 6.8 and 6.5.3. Those requirements include review of changes for unreviewed safety questions in accordance with the provisions of 10CFR50.59. The requirements of 10CFR50.59 include a review of the evaluated change for impact on the probability of an accident previously evaluated. The requirements of 10CFR50.59 prevent any evaluated change which increases the probability or consequences of an accident previously evaluated from being made without prior NRC approval. These changes, therefore constitute an administrative revision only.

   Therefore, this change does not significantly increase the probability or consequences of a previously evaluated accident.

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

   The proposed change will not involve any plant design change. The proposed change will not alter the operation of the plant or the manner in which it is operated (i.e., administrative only). Therefore, the proposed change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The information proposed for relocation will be adequately controlled via the administrative requirements of Technical Specification 6.8 and 6.5.3. Those requirements include review of changes for unreviewed safety questions in accordance with the provisions of 10CFR50.59. The requirements of 10CFR50.59 include a review of the evaluated change for impact on the probability of an accident previously evaluated. The requirements of 10CFR50.59 prevent any evaluated change which increases the probability or consequences of an accident previously evaluated from being made without prior NRC approval. These changes, therefore constitute an administrative revision only.

   Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This proposed change relocates accident monitoring instrumentation that is not designated as R. G. 1.97 Category I or Type A from the Technical Specifications to plant administrative control. The proposed change will have no adverse impact on the reactor coolant system pressure boundary nor will any other system protective boundary or safety limit be affected. Accident monitoring instrumentation that is designated as Category I or Type A are being retained in the technical specifications.

   The information proposed for relocation will be adequately controlled via the administrative requirements of Technical Specification 6.8 and 6.5.3. Those requirements include review of changes for unreviewed safety questions in accordance with the provisions of 10CFR50.59. The requirements of 10CFR50.59 include a review of the evaluated change for impact on the probability of an accident previously evaluated. The requirements of 10CFR50.59 prevent any evaluated change which increases the probability or consequences of an accident previously evaluated from being made without prior NRC approval. These changes, therefore constitute an administrative revision only.

   Therefore, the proposed change will not involve a significant reduction in a margin of safety.

Based on the above evaluation in accordance with 10CFR50.92(c), Entergy Operations, Inc. has concluded that operation of the facility in accordance with the proposed amendment involves no significant hazards consideration.