DOCKET FILE

Docket No. 50-416 August 13, 1992

Mr. William T. Cottle
Vice President, Operations GGNS
Entergy Operations, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

Dear Mr. Cottle:

SUBJECT: CORRECTION TO AMENDMENT NO. 97 TO FACILITY OPERATING LICENSE NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M80700)

On May 20, 1992, the Commission issued Amendment No. 97 to Facility Operating License No. NPF-29 to Grand Gulf Nuclear Station, Unit 1. The amendment revised the Technical Specifications (TS) to extend the surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for instrumentation supporting the Emergency Core Cooling System (ECCS), Control Rod Block Function (CRBF), and Isolation Actuation Instrumentation. Editorial changes were also made to the existing TS to clarify the intent of the revisions.

Correction is being made to Bases pages B 3/4 3-1 through B 3/4 3-4. TS page B 3/4 3-1 should not have had the amendment number typed on the bottom of the page. It was an overleaf page and contained no changes. In addition, due to repagination, Section 3/4.3.7.5, Accident Monitoring Instrumentation, was accidently deleted from the bottom of page B 3/4 3-4. Also, none of the Bases pages contained any vertical lines indicating the areas of change. We are issuing these corrected pages to maintain document completeness. We apologize for any inconvenience these errors may have caused you.

Sincerely,
ORIGINAL SIGNED BY
Paul W. O'Connor, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.
b. Preserve the integrity of the reactor coolant system.
c. Minimize the energy which must be absorbed following a loss-of-coolant accident, and
d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P, "Technical Specification Improvement Analyses for BWR Reactor Protection System," as approved by the NRC and documented in the SER (letter T. A. Pickens from A. Thadani dated July 15, 1987). The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

ISOLATION ACTUATION INSTRUMENTATION (Continued)

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with: (1) NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (letter to D. N. Grace from C. E. Rossi dated January 6, 1989) and (2) NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (letter to S. D. Floyd from C. E. Rossi dated June 18, 1990).

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 10 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 10 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 10 second delay. It follows that checking the valve speeds and the 10 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Negative barometric pressure fluctuations are accounted for in the trip setpoints and allowable values specified for drywell pressure-high. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, Parts 1 and 2, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)" as approved by the NRC and documented in the NRC Safety Evaluation Reports (letter to D. N. Grace from A. C. Thadani dated December 9, 1988 (Part 1) and letter to D. N. Grace from C. E. Rossi dated December 9, 1988 (Part 2)).

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION (Continued)

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram recirculation pump trip (ATWS-RPT) system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event has been evaluated in General Electric Company report NEDC-32408 dated March 1987. The results of the analysis show that the Grand Gulf ATWS-RPT design provides adequate protection for these events in which the normal scram paths fail.

The ATWS-RPT provides fully redundant trip of the recirculation pump motors so that the pumps coast down to zero speed. This trip function reduces core flow creating steam voids in the core, thereby decreasing power generation and limiting any power or pressure excursions. The Grand Gulf ATWS-RPT design provides compliance with the requirements of the NRC ATWS Rule 10CFR50.62.

The ATWS-RPT and Alternate Rod Insertion (ARI) system use common setpoints and trip channels (transmitters and trip systems). Therefore, the ARI trip function and the RPT trip function will be initiated simultaneously. The instrumentation setpoints for the RPV pressure and water level trip channels are established such that the normal scram paths for these variables would already be initiated.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a closure sensor for each of two turbine stop valves provides input to one EOC-RPT system; a closure sensor from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room. The automatic bypass setpoint is feedwater temperature dependent due to the subcooling changes that affect the turbine first-stage pressure-reactor power relationship. For RATED THERMAL POWER operation with feedwater temperature greater than or equal to 420°F, an allowable setpoint of < 26.9% of control valve wide open turbine first-stage pressure is provided for the bypass function. This setpoint is also applicable to operation at less than RATED THERMAL POWER with the correspondingly lower feedwater temperature. The allowable setpoint is reduced to < 22.5% of control valve wide open turbine first-stage pressure for RATED THERMAL POWER operation with a feedwater temperature between 370°F and 420°F. Similarly, the reduced setpoint is applicable to operation at less than RATED THERMAL POWER with the corresponding lower feedwater temperature.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms. Included in this time are: the response time of the sensor, the response time of the system logic and the breaker interruption time. Breaker interruption time includes both breaker response time and the manufacturer's design arc suppression time of 12 ms.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater then the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will results in a control rod block.

The OPERABILITY of the control rod block instrumentation in OPERATIONAL CONDITION 5 is to provide diversity of rod block protection to the one-rod-out interlock.

Specified surveillance intervals have been determined in accordance with NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation" as approved by the NRC and documented in the NRC Safety Evaluation Report (letter to D. N. Grace from C. E. Rossi dated September 22, 1988).

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION (Continued)

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of NUREG-0737, "Clarification of TMI Action Plan Requirements," November, 1980.

3/4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION AND CONTROLS

The OPERABILITY of the remote shutdown system instrumentation and controls ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR 50.

3/4.3.7.5 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess important variables following an accident. This capability is consistent with the recommendations of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".