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 DENTON,H.R. Office of Nuclear Reactor Regulation, Director
 STOLZ,J.F. Operating Reactors Branch 4

SUBJECT: Forwards supplemental response to Generic Ltr 83-37 re Tech Specs for items scheduled for implementation after 811231 & request for revs to Tech Specs. Request resulted from Commission action & should be exempt from fees.

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NOTES: 1 1

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HAL B. TUCKER
VICE PRESIDENT
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October 8, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. John F. Stolz, Chief
Operating Reactors Branch No. 4

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

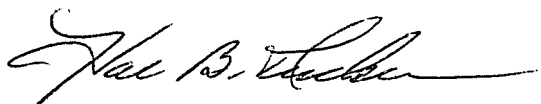
By letter dated November 1, 1983, the NRC provided Generic Letter 83-37 concerning Technical Specifications for items scheduled for implementation after December 31, 1981. My letters dated December 2, 1983 and December 14, 1983 provided initial response to this Staff request.

Attachment 1 provides detailed responses to the eleven items contained in Enclosure 1 to the Generic Letter. As a result of our review, Technical Specifications were developed and are included in Attachment 2.

Please note that containment level monitors LT-120, 113, 3P and 112 proposed under items 2.a and 2.b in Table 3.5.6.1 (Attachment 2) have not been installed in Oconee Units 1 and 2. However, the installation of these monitors is scheduled for completion during the upcoming refueling outages.

Inasmuch as the request for these proposed changes resulted from a written Commission request, and it has only minor safety significance and may, pursuant to 10 CFR 170.21, be exempt from fees, Duke considers that no license fees are justified.

Very truly yours,



Hal B. Tucker

MAH:slb

Attachment

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Mr. Harold R. Denton, Director

October 8, 1984

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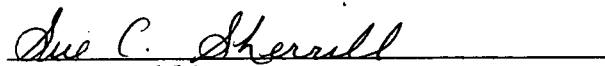
Mr. Harold R. Denton, Director
October 8, 1984
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HAL B. TUCKER, being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, DPR-37, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.



Hal B. Tucker, Vice President

Subscribed and sworn to before me the 8th day of October, 1984.


Notary Public

My Commission Expires:

September 20, 1989

Attachment 1
Duke Power Company
Oconee Nuclear Station
Response to NRC Generic Letter 83-37
NUREG-0737 Technical Specifications
(Items scheduled after 12/31/81)

1. Reactor Coolant System Vents (II.B.1)

Clarification (11) of this NUREG item states:

Provisions to test for operability of the reactor coolant vent system should be a part of the design. Testing should be performed in accordance with subsection IWV of Section XI of the ASME Code for Category B valves.

Oconee Technical Specifications presently require inservice testing of ASME pumps and valves in accordance with Section XI.

The RCS vent valves were included in the inservice test program submitted for NRC review by letters dated November 19, 1982 and October 10, 1983.

Thus, it is considered that the original requirement of this NUREG item is met and no changes are necessary.

2. Post-Accident Sampling (II.B.3)

Duke will establish an administrative program which includes training of personnel, procedures for sampling and analysis, provisions for maintenance of sampling and analysis equipment which will be referenced in the Administrative Controls section of the Technical Specifications as proposed Technical Specification 6.4.3 in Attachment 2.

3. Long Term Auxiliary Feedwater System Evaluation (II.E.1.1)

Existing Oconee Technical Specifications 3.4 and 4.9 effectively govern operability and surveillance of the emergency feedwater system. As such, no changes are deemed necessary.

4. Noble Gas Effluent Monitors (II.F.1.1)

The NRC model Technical Specification has been reviewed and an appropriate proposal for Oconee, Technical Specification 3.5.6.1, is included in Attachment 2.

5. Sampling and Analysis of Plant Effluents (II.F.1.2)

A response consistent with that provided to Item 2 is provided as proposed Technical Specifications 6.4.4 in Attachment 2.

6. Containment High-Range Radiation Monitor (II.F.1.3)

The NRC model Technical Specification has been reviewed and an appropriate proposal for Oconee as Technical Specification 3.5.6 is included in Attachment 2.

7. Containment Pressure Monitor (II.F.1.4)

The NRC model Technical Specification has been reviewed and an appropriate proposal for Oconee as Technical Specification 3.5.6 is included in Attachment 2.

8. Containment Water Level Monitor (II.F.1.5)

The NRC model Technical Specification has been reviewed and an appropriate proposal for Oconee as Technical Specification 3.5.6 is included in Attachment 2.

9. Containment Hydrogen Monitor (II.F.1.6)

The NRC model Technical Specification has been reviewed and an appropriate proposal for Oconee as Technical Specification 3.5.6 is included in Attachment 2.

10. Instrumentation for Detection of Inadequate Core Cooling (II.F.2)

Duke has reviewed the NRC model Technical Specifications provided for this item and considers it premature to be developing limiting conditions for operation and surveillance for a system which is yet to be designed, procured, or installed. As such, no proposal relative to this item will be submitted at this time.

11. Control Room Habitability Requirements (III.D.3.4)

This item is currently under active review with the Staff. Upon completion of committed plant modifications by Duke and NRC acceptance of the final design, Duke will determine appropriate Technical Specifications for this item.

Duke Power Company
Oconee Nuclear Station
Attachment 2
Proposed Technical Specification Revision

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3.5.6 Accident Monitoring Instrumentation

Applicability

Applies to accident monitoring instrumentation.

Objective

To ensure that sufficient information is available on selected plant parameters to monitor and assess such parameters following an accident.

Specifications

- 3.5.6.1 The accident monitoring instrumentation shown in Table 3.5.6.1 shall be operable per applicability indicated in the Table.
- 3.5.6.2 In the event that the number of accident monitoring instrumentation channels falls below the limit given in Table 3.5.6.1, exert best efforts to return the instrumentation to operable status within 7 days of the event or a report shall be submitted to the Commission within the next 30 days outlining the cause of inoperability and the plans and schedule for restoring the instrumentation to operable status.
- 3.5.6.3 The provisions of Technical Specification 3.0 do not apply.

Bases

The operability of the accident monitoring instrumentation for accident conditions as appropriate ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. The operability and use of this instrumentation is consistent with the requirement of General Design Criterion 14 of Appendix A to 10 CFR Part 50.

Table 3.5.6.1
ACCIDENT MONITORING INSTRUMENTATION

<u>Instrument</u>	<u>Minimum Operable Channels</u>	<u>Applicability</u>
1. Containment Pressure Monitor (PT-230, -231)	1 of 2	(a)
2. Containment Water Level Monitors		(a)
a) Wide Range (LT-90, 91)	1 of 2	
b) Normal Sump (LT-120, 113)	1 of 2	
c) Emergency Sump (LT-3P, 112)	1 of 2	
3. Noble Gas Effluent Monitor (RIA-56)	1 of 2	(a)
4. Containment High-Range Radiation Monitor (RIA-57, -58)	1 of 2	(a)
5. Containment Hydrogen Monitor (MT-80, 81)	1 of 2	(a)

(a) At all times except for cold shutdown and refueling outages.

Table 4.1-1 (CONTINUED)

<u>Channel Description</u>	<u>Check</u>	<u>Test</u>	<u>Calibrate</u>	<u>Remarks</u>
49. Emergency Feedwater Flow Indicators	MO	NA	RF	
50. PORV and Safety Valve Position Indicators	MO	NA	RF	
51. RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	MO	RF	
52. RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
a) Control Oil Pressure Switches	NA	MO	RF	
b) Discharge Pressure Switches	NA	MO	RF	
53. Emergency Feedwater Initiation Circuits				
a) Control Oil Pressure Switches	NA	MO	RF	
b) Discharge Pressure Switches	NA	MO	RF	
54. Containment High Range Radiation Monitor (RIA-57,58)	NA	MO	RF	TMI Item II.F.1.3

Table 4.1-1 (CONTINUED)

<u>Channel Description</u>	<u>Check</u>	<u>Test</u>	<u>Calibrate</u>	<u>Remarks</u>
55. Containment Pressure Monitor (PT-230,231)	MO	NA	AN	TMI Item II.F.1.4
56. Containment Water Level Monitors				TMI Item II.F.1.5
a) Wide Range (LT-90, 91)	MO	NA	RF	
b) Normal Sump (LT-120, 113)	MO	NA	RF	
c) Emergency Sump (LT-3P, 112)	MO	NA	RF	
57. Containment Hydrogen Monitor (MT-80,-81)	MO	NA	AN	TMI Item II.F.1.6
58. Noble Gas Effluent Monitor (RIA-56)	NA	MO	AN	TMI Item II.F.1.1

ES - Each Shift
 DA - Daily
 WE - Weekly
 MO - Monthly

QU - Quarterly
 AN - Annually
 PS - Prior to startup, if not performed previous week
 NA - Not Applicable
 RF - Refueling Outage

- 6.4.3 The station shall have a program that ensures the capability to obtain and analyze reactor coolant and containment atmosphere samples under accident conditions which includes training of personnel, procedures for sampling and analysis, and provisions for testing and required maintenance of sampling and analysis equipment.
- 6.4.4 The station shall have a program that ensures the capability to collect and analyze or measure representative samples of radioactive iodines and particulates in plant gaseous effluents during and following an accident which includes training of personnel, procedures for sampling and analysis, and provisions for testing and required maintenance of sampling and analysis equipment.

Duke Power Company
Oconee Nuclear Station
Attachment 3
No Significant Hazards Consideration Evaluation

No Significant Hazards Consideration Evaluation

Duke Power Company (Duke) has made the determination that this amendment request involves a No Significant Hazards Consideration by applying the standards established by the Commission's regulations in 10 CFR 50.92. This ensures that 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 margin of safety.

The Commission has provided guidance concerning the application of these standards by providing certain examples (48 FR 14870). Example (ii) of types of amendments considered not likely to involve significant hazards consideration is applicable to this amendment request. This specific example involves amendment requests that are considered to be a change that constitutes an additional limitation, or control not presently included in the technical specifications; for example, a more stringent surveillance requirement.

The proposed Technical Specification amendment addressed in this submittal has been determined by Duke to be a change in conformance to the regulations for additional limitation and control. Briefly, the proposed amendment pertain to NUREG-0737 items implemented after December 31, 1981, in response to the requirements of Generic Letter 83-37, and results in very minor changes to facility operations clearly in keeping with the regulations.

Duke has determined, based on consideration that the requested amendment is a conformatory change in nature, that the revision does not involve a significant increase in the probability or consequences of accidents previously considered, nor create the possibility of a new or different kind of accident, and will not involve a significant decrease in a safety margin. Therefore, Duke concludes that there is no significant hazards consideration involved in this amendment request.