Duke Power Company

Oconee Nuclear Station

Attachment 2

Proposed Technical Specifications

Remove Pages	Insert Pages
iii through v	iii through v
vi	vi
3.1-24	
	3.5-44 through -46
3.15-1	3.15-1
	3.15-2
4.1-8	4.1-8
	4.1-8a
4.1-9	4.1-9
4.1-12	4.1-12

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Section		Page
3.1.1	Operational Component	3.1-1
3.1.2	Pressurization, Heatup and Cooldown Limitations	3.1-3
3.1.3	Minimum Conditions for Criticality	3.1-8
3.1.4	Reactor Coolant System Activity	3.1-10
3.1.5	Chemistry	3.1-12
3.1.6	Leakage	3.1-14
3.1.7	Moderator Temperature Coefficient of Reactivity	3.1-17
3.1.8	(Intentionally Blank)	3.1-19
3.1.9	Low Power Physics Testing Restrictions	3.1-20
3.1.10	Control Rod Operation	3.1-21
3.1.11	Shutdown Margin	3.1-23
3.2	HIGH PRESSURE INJECTION AND CHEMICAL ADDITION SYSTEMS	3.2-1
3.3	EMERGENCY CORE COOLING, REACTOR BUILDING COOLING, REACTOR BUILDING SPRAY AND LOW PRESSURE SERVICE WATER SYSTEMS	3.3-1
3.4	SECONDARY SYSTEM DECAY HEAT REMOVAL	3.4-1
3.5	INSTRUMENTATION SYSTEMS	3.5-1
3.5.1	Operational Safety Instrumentation	3.5-1
3.5.2	Control Rod Group and Power Distribution Limits	3.5-6
3.5.3	Engineered Safety Features Protective System Actuation Setpoints	3.5-31
3.5.4	Incore Instrumentation	3.5-33
3.5.5	Radioactive Effluent Monitoring Instrumentation	3.5-37
3.5.6	Accident Monitoring Instrumentation	3.5-44
3.6	REACTOR BUILDING	3.6-1
3.7	AUXILIARY ELECTRICAL SYSTEMS	3.7-1
3.8	FUEL LOADING AND REFUELING	3.8-1
3.9	RADIOACTIVE LIQUID EFFLUENTS	3.9-1

Section		Page
3.10	RADIOACTIVE GASEOUS EFFLUENTS	3.10-1
3.11	SOLID RADIOACTIVE WASTE	3.11-1
3.12	REACTOR BUILDING POLAR CRANE AND AUXILIARY HOIST	3.12-1
3.13	SECONDARY SYSTEM ACTIVITY	3.13-1
3.14	SNUBBERS	3.14-1
3.15	CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM AND PENETRATION ROOM VENTILATION SYSTEMS	3.15-1
3.16	HYDROGEN PURGE SYSTEM	3.16-1
3.17	FIRE PROTECTION AND DETECTION SYSTEMS	3.17-1
4	SURVEILLANCE REQUIREMENTS	4.0-1
4.0	SURVEILLANCE STANDARDS	4.0-1
4.1	OPERATIONAL SAFETY REVIEW	4.1-1
4.2	STRUCTURAL INTEGRITY OF ASME CODE CLASS 1, 2 AND 3 COMPONENTS	4.2-1
4.3	TESTING FOLLOWING OPENING OF SYSTEM	4.3-1
4.4	REACTOR BUILDING	4.4-1
4.4.1	Containment Leakage Tests	4.4-1
4.4.2	Structural Integrity	4.4-14
4.4.3	Hydrogen Purge System	4.4-17
4.4.4	Reactor Building Purge System	4.4-20
4.5	EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEMS PERIODIC TESTING	4.5-1
4.5.1	Emergency Core Cooling Systems	4.5-1
4.5.2	Reactor Building Cooling Systems	4.5-6
4.5.3	Penetration Room Ventilation System	4.5-10
4.5.4	Low Pressure Injection System Leakage	4.5-12
4.6	EMERGENCY POWER PERIODIC TESTING	4.6-1
4.7	REACTOR CONTROL ROD SYSTEM TESTS	4.7-1
4.7.1	Control Rod Trip Insertion Time	4.7-1
4.7.2	Control Rod Program Verification	4.7-2
4.8	MAIN STEAM STOP VALVES	4.8-1

Section		Page
4.9	EMERGENCY FEEDWATER PUMP AND VALVE PERIODIC TESTING	4.9-1
4.10	REACTIVITY ANOMALIES	4.10-1
4.11	RADIOLOGICAL ENVIRONMENTAL MONITORING	4.11-1
4.12	CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM	4.12-1
	(INTENTIONALLY BLANK)	4.13-1
4.14	REACTOR BUILDING PURGE FILTERS AND SPENT FUEL POOL VENTILATION SYSTEM	4.14-1
4.15	IODINE RADIATION MONITORING FILTERS	4.15-1
4.16	RADIOACTIVE MATERIALS SOURCES	4.16-1
4.17	STEAM GENERATOR TUBING SURVEILLANCE	4.17-1
4.18	SNUBBERS	4.18-1
4.19	FIRE PROTECTION AND DETECTION SYSTEM	4.19-1
4.20	DELETED PER AMENDMENTS 109, 109, and 106	
4.21	DOSE CALCULATIONS	4.21-1
5	DESIGN FEATURES	5.1-1
5.1	SITE	5.1-1
5.2	CONTAINMENT	5.2-1
5.3	REACTOR	5.3-1
5.4	NEW AND SPENT FUEL STORAGE FACILITIES	5.4-1
6	ADMINISTRATIVE CONTROLS	6.1-1
6.1	ORGANIZATION, REVIEW, AND AUDIT	6.1-1
6.1.1	Organization	6.1-1
6.1.2	Technical Review and Control	6.1-2
6.1.3	Nuclear Safety Review Board	6.1-3a
6.2	ACTION TO BE TAKEN IN THE EVENT OF A REPORTABLE OCCURRENCE	6.2-1
6.3	ACTION TO BE TAKEN IN THE EVENT A SAFETY LIMIT IS EXCEEDED	6.3-1
6.4	STATION OPERATING PROCEDURES	6.4-1

LIST OF TABLES

Table No.		Page
2.3-1	Reactor Protective System Trip Setting Limits - Units 1,2 and 3	2.3-7
3.5.1-1	Instruments Operating Conditions	3.5-4
3.5-1	Quadrant Power Tilt Limits	3.5-14
3.5.5-1	Liquid Effluent Monitoring Instrumentation Operating Conditions	3.5-39
3.5.5-2	Gaseous Process and Effluent Monitoring Instrumentation Operating Conditions	3.5-41
3.5.6-1	Accident Monitoring Instrumentation	3.5-45
3.7-1	Operability Requirements for the Emergency Power Switching Logic Circuits	3.7-14
3.17-1	Fire Protection & Detection Systems	3.17-5
4.1-1	Instrument Surveillance Requirements	4.1-3
4.1-2	Minimum Equipment Test Frequency	4.1-9
4.1-3	Minimum Sampling Frequency and Analysis Program	4.1-10
4.1-4	Radioactive Effluent Monitoring Instrumentation Surveillance Requirements	4.1-16
4.4-1	List of Penetrations with 10CFR50 Appendix J Test Requirements	4.4-6
4.11-1	Radiological Environmental Monitoring Program	4.11-3
4.11-2	Maximum Values for the Lower Limits of Detection (LLD)	4.11-5
4.11-3	Reporting Levels for Radioactivity Concentrations in Environmental Samples	4.11-8
4.17-1	Steam Generator Tube Inspection	4.17-6
6.1-1	Minimum Operating Shift Requirements with Fuel in Three Reactor Vessels	6.1-6

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. The provisions of Technical Specification 3.0 do not apply.

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 Column A; operation shall be limited as specified in Column B.

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.

RCS subcooled margin is directly indicated in the control room. Core subcooled margin is indicated on both ICC plasma displays, the OAC video, and a digital control board meter. Loop A subcooled margin is indicated on one ICC plasma display, the OAC video, and a digital control board meter. Loop B subcooled margin is indicated on the other ICC plasma display, the OAC video, and a digital control board meter. The OAC video and the digital control board meters are redundant displays of the same signal.

The operability requirements of the Reactor Coolant System subcooling margin monitors ensures that sufficient information is available to the operators to provide prompt recognition of saturated conditions in the primary coolant system and advanced warning of the approach to inadequate core cooling. Guidance for these requirements was provided by the NRC letter of July 2, 1980, and derived from the implementation of the TMI-2 lessons learned program.

Temperature indications from all 24 qualified core exit thermcouples can be displayed on the OAC. 12 qualified core exit thermcouples per train will input to each train of process electronics and can be displayed on the respective ICC plasma display.

Table 3.5.6-1
ACCIDENT MONITORING INSTRUMENTATION

	Instrument	(A) Required Operable Channels	(B) Action	(C) Applicability
1.	Containment Pressure Monitor (PT-230, -231)	2 of 2	1	Above hot shutdown
2.	Containment Water Level Monitor Wide Range (LT-90, -91)	2 of 2	2	Above hot shutdown
3.	Containment High-Range Radiation Monitor (RIA-57, -58)	2 of 2	2	Above hot shutdown
4.	Containment Hydrogen Monitor (MT-80, -81)	2 of 2	2	Above hot shutdown
5.	Wide Range Hot Leg Level (RC-LT0123, RC-LT0124)	2 of 2	3	Above hot shutdown
6.	Reactor Vessel Head Level (RC-LT0125, RC-LT0126)	2 of 2	3	Above hot shutdown
7.	Qualified Core Exit Thermocouple Trains	2 of 2 (a)	2	Above hot shutdown
8.	Subcooling Monitors	2 (b)	4	When RCS temperature is >300°F

Table 3.5.6-1 (CONTINUED)
ACCIDENT MONITORING INSTRUMENTATION

ACTIONS

Action 1: If one channel is inoperable, the channel shall be restored to operable status within 7 days, or the unit shall be in hot shutdown within the next 12 hours.

If two channels are inoperable, at least one channel shall be restored to operable status within 48 hours, or the unit shall be in hot shutdown within the next 12 hours.

Action 2: If one channel is inoperable, the channel shall be restored to operable status within 30 days, or the unit shall be in hot shutdown within the next 12 hours.

If two channels are inoperable, at least one channel shall be restored to operable status within 48 hours, or the unit shall be in hot shutdown within the next 12 hours.

Action 3: If one channel is inoperable, the channel shall be restored to operable status within 7 days, or a report shall be submitted to the Commission within the next 30 days outlining the cause of the inoperability and the plans and schedule for restoring the channel to operable status.

If two channels are inoperable, at least one channel shall be restored to operable status within 7 days, or the unit shall be in hot shutdown within the next 12 hours.

Action 4: If one of the required channels is inoperable, at least one channel shall be restored to operable status within 30 days or the unit shall be in hot shutdown within the next 12 hours and below 300°F within the next 24 hours.

If two of the required channels are inoperable, at least one channel shall be restored to operable status within 48 hours or the unit shall be in hot shutdown within the next 12 hours and below 300°F within the next 24 hours.

NOTES

(a) 5 of 12 qualified core exit thermocouples must be operable per train for a train to be considered operable.

(b) Operable subcooling margin monitors must consist of:

1) One direct indication for 1 of 2 RCS hot legs and one direct indication for the core; or

2) One direct indication for each RCS hot leg.

3.15 Control Room Pressurization and Filtering System and Penetration Room Ventilation System

Applicability

Applies to the Unit 1 and 2, and Unit 3 control room pressurization and filtering systems and the penetration room ventilation system.

Objective

To define the conditions necessary to assure operability of the control room pressurization and filtering system and the immediate availability of the penetration room ventilation systems.

Specification

3.15.1 Penetration Room Ventilation Systems

a. Two trains of the penetration room ventilation systems shall be operable at all times when containment integrity is required or the reactor shall be shutdown within 12 hours with the following exception:

(1) If one of two trains of a penetration room ventilation system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days provided that all active components of the other train of the penetration room ventilation system shall be demonstrated to be operable within 24 hours and daily thereafter.

3.15.2 Control Room Pressurization and Filtering Systems

a. With the reactor above hot shutdown conditions both outside air booster fans shall be operable.

(1) If one outside air booster fan is inoperable, restore the inoperable fan to operable status within 72 hours, or the unit shall be in hot shutdown within the next 12 hours.

(2) If both outside air booster fans are inoperable, restore at least one inoperable fan to operable status within 24 hours or the unit shall be in hot shutdown within the next 12 hours.

b. With the reactor above hot shutdown conditions and both outside air booster fans operable, the control room pressurization and filtering systems shall be capable of maintaining a positive pressure within the control room.

   (1) If the above requirements of Specification 3.15.2.b are not met within 30 days, the unit shall be in hot shutdown within the next 12 hours.

c. With the reactor above hot shutdown conditions, both filter trains shall be operable.

   (1) If one filter train is inoperable, restore the inoperable filter train to operable status within 72 hours, or the unit shall be in hot shutdown within the next 12 hours.

   (2) If both filter trains are inoperable, restore one inoperable filter train to operable status within 24 hours or the unit shall be in hot shutdown within the next 12 hours.

d. The provisions of Specification 3.0 do not apply.

Bases

A single train of reactor building penetration room ventilation equipment retains full capacity to control and minimize the release of radioactive materials from the reactor building to the environment in post-accident conditions.

The control room pressurization and filtering system is comprised of two separate outside air booster fans with prefilter/HEPA/carbon filter trains, two redundant control room air handling unit fans, and associated ductwork. The system is designed to protect the control room operators from the effects of accidental release of radioactive effluents or toxic gases in the Turbine or Auxiliary Building.

Protection is provided by pressurizing the control room with filtered outside air to prevent inleakage of radioactive effluents or toxic gases from the Turbine or Auxiliary Building only. Specification 3.15.2.b applies to all instances where the reactor is above hot shutdown and the system is judged incapable of maintaining the control room at a positive pressure or, if during refueling frequency testing per Specification 4.12.1.b the system is demonstrated to be incapable of maintaining the control room at a positive pressure.

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
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)

Channel Description	Check	Test	Calibrate	Remarks
55. Containment Pressure Monitor (PT-230, 231)	MO	NA	AN	TMI Item II.F.1.4
56. Containment Water Level Monitor-Wide Range (LT-90, -91)	MO	NA	RF	TMI Item II.F.1.5
57. Containment Hydrogen Monitor (MT-80,-81)	NA	MO	AN	TMI Item II.F.1.6
58. Wide Range Hot Leg Level	NA	RF	RF	
59. Reactor Vessel Head Level	NA	RF	RF	
60. Core Exit Thermocouples	MO	NA	RF	
61. Subcooling Monitors	MO	RF	RF	

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

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rod Movement (1)	Movement of Each Rod	Monthly
2.	Pressurizer Safety Valves	Setpoint	Each Refueling (4)
3.	Main Steam Safety Valves	Setpoint	Each Refueling (4)
4.	Refueling System Interlocks	Functional	Prior to Refueling
5.	Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly
6.	Reactor Coolant System Leakage (2)	Evaluate	Daily
7.	Condenser Cooling Water System Gravity Flow Test	Functional	Each Refueling
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
9.	Spent Fuel Cooling System	Functional	Prior to Refueling
10.	High Pressure and Low Pressure Injection System (3)	Vent Pump Casings	Monthly and Prior to Testing
11.	Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	Each Refueling

(1) Applicable only when the reactor is critical.

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

(3) Operating pumps excluded.

(4) Number of safety valves to be tested each refueling shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

4.12 CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM

Applicability

Applies to control room pressurization and filtering system components

Objective

To verify that these systems and components will be able to perform their design functions.

Specification

4.12.1 Operating Tests

a. Control room outside air booster fan system tests shall be performed quarterly. These tests shall consist of an external visual inspection, a flow measurement for each unit and pressure drop measurements across each filter bank. Pressure drop across pre-filter shall not exceed 1 inch H_2O and pressure drop across HEPA shall not exceed 2 inches H_2O. Fan motors shall be operated continuously for at least one hour, and all louvers shall be proven operable.

b. On a refueling frequency, verify the system maintains the control room at a positive pressure with both outside air booster fans on during system operation.

4.12.2 Filter Tests

On a refueling frequency, for the Unit 1 and 2 and the Unit 3 control room an in-place leakage test using DOP on HEPA units and Freon-112 (or equivalent) on carbon units shall be performed at design flow on each filter train. Removal of 99.5 percent DOP by each entire HEPA filter unit and removal of 99.0 percent Freon-112 (or equivalent) by each entire carbon adsorber unit shall constitute acceptance performance. These tests must also be performed after any maintenance which may affect the structural integrity of either the filtration system units or of the housing.

Bases

The purpose of the control room pressurization filtering system is to protect the control room operators from the effects of accidental release of radioactive effluents or toxic gases in the Turbine Building or Auxiliary Building only. The system is designed with two 50 percent capacity filter trains each of which consists of a prefilter, high efficiency particulate filters, carbon filters, booster fans, air handling unit fans, and associated ductwork to pressurize the control room with outside air.

Since these systems are not normally operated, a periodic test is required to insure their operability when needed. Quarterly testing of this system will show that the system is available.

Refueling frequency testing of the installed carbon adsorber stage and absolute filters will verify the leak integrity of the cleanup system. Refueling frequency testing will also verify the ability of the system to maintain the control room at a positive pressure to minimize infiltration of hazardous effluents.

Duke Power Company

Oconee Nuclear Station

Attachment 3

Technical Justification

Technical Justification

The following list of correspondence provides the necessary information for an adequate technical review of Duke's response to GL 83-37. Given the short period of time for development of this proposal, a more in depth summary of the items is not possible. However, a brief summary of the technical highlights of each issue is provided.

Duke Power Letters to NRC	NRC Letters to Duke Power
December 2, 1983	November 1, 1983
December 14, 1983	October 31, 1985
October 8, 1984	November 25, 1985
August 27, 1985	March 31, 1987
January 30, 1986	November 19, 1987
June 27, 1986	November 15, 1988
August 13, 1986	
September 19, 1986	
January 6, 1988	
January 18, 1988	
March 15, 1988	
May 13, 1988	
September 16, 1988	
December 29, 1988	

GL 83-37 Item II.B.1
Reactor Coolant System (RCS) High Point Vents

RCS high point vents are provided to exhaust non-condensible gases and/or steam that could inhibit natural circulation core cooling following an accident beyond the design basis of Oconee. The NRC staff has previously approved relocation of Technical Specifications regarding RCS high point vents (T. E. Murley, NRC to W. S. Wilgus, BWOG May 6, 1988) based upon the Commission's Interim Policy Statement on Technical Specification Improvements (52 FR 3788). As such, with verbal NRC staff approval on May 11, 1989 Duke is withdrawing proposals for inclusion of this item within Technical Specifications. Commitments with regard to operability and surveillance of the high point vents will be relocated to the FSAR (Attachment 1).

GL 83-37 Item II.F.1.1
Noble Gas Effluent Monitor (RIA-56)

The noble gas effluent monitor (RIA-56) is used for detection of significant releases and dose assessments. No specific operator actions have been identified as part of any design basis event which utilizes this instrument. As such, with verbal NRC staff approval on May 11, 1989 Duke is withdrawing proposals for inclusion of this item within Technical Specifications. Commitments with regard to operability and surveillance of the noble gas effluent monitor will be relocated to the FSAR (Attachment 1).

GL 83-37 Item II.F.1.3
Containment High Range Radiation Monitors (RIA-57,-58)

The principle function of RIA-57 and RIA-58 is to monitor significant post accident radiation levels within containment. No operator actions utilizing these instruments as part of a primary success path have been identified. Further, a review of the Oconee 3 PRA indicates that these instruments are not utilized in any risk-significant events. However, consistent with verbal NRC staff approval on May 11, 1989 Duke is providing Technical Specifications including shutdown requirements (Attachment 2).

GL 83-37 Item II.F.1.4
Containment Pressure Monitor

The principle use of the containment pressure monitor is to confirm that containment heat removal systems are causing a containment pressure decrease following an energy release. Thus it permits backup verification of heat removal system performance. Failure to control pressure usually means that all heat removal systems have failed (beyond Design Basis) or the energy input is excessive and that systems to limit energy have failed in some fashion (high volume H_2, continued secondary steam input, failure to trip the reactor, etc.). It is not considered a R.G.1.97 Type A variable as no operator actions are required. However, a review of the Oconee 3 PRA indicates that this instrument/variable is utilized is risk-significant events at Oconee. Consistent with verbal NRC staff approval on May 11, 1989 Duke is providing Technical Specifications including shutdown requirements (Attachment 2).

GL 83-37 Item II.F.1.5
Containment Water Level Monitor Wide Range

The principle function of the wide range containment water level monitor is to confirm reactor building water level prior to swapping suction from the BWST to the reactor building sump for certain events which are beyond the design basis for Oconee. A review of the Oconee 3 PRA indicates that this instrument is not utilized in any risk-significant events.

The principle instrument used to determine when to manually swap LPI pump suction from the BWST to the containment sump is BWST level instrumentation. BWST level instrumentation is currently required to be operable by Specification 3.3.4. However, consistent with verbal NRC staff approval on May 11, 1989 Duke is providing Technical Specifications including shutdown requirements. (Attachment 2)

GL 83-37 Item II.F.1.6
Containment Hydrogen Monitor

Containment hydrogen monitors are provided to detect excessive hydrogen concentrations within containment following an accident beyond the design basis for Oconee. Although emergency operating procedures include references to the hydrogen monitors, operator actions required by the procedures are provided for mitigation of beyond design basis events (including termination of recombiner operation). For design basis events, initiation of hydrogen mitigation systems is based upon time after the event; not hydrogen monitor indications. However, consistent with verbal NRC staff approval on May 11, 1989 Duke is providing Technical Specifications including shutdown requirements (Attachment 2).

GL 83-37 Item II.F.2
Reactor Vessel Level Indication System (RVLIS)

Inventory tracking systems are post-TMI 2 concepts that will provide information for any condition of partial voiding, but are primarily intended for unusual events where extraordinary measures (such as secondary blowdown, or loop or head venting) may be needed. Credit for inventory tracking systems is not needed for Design Basis Accidents.

The only DBA which would result in Inadequate Core Cooling (ICC) conditions is a large break LOCA. Information from RVLIS is not essential and lack of such information will not prevent the operator from responding to and mitigating the consequences of any DBA identified in the Oconee FSAR. Rather, information from the subcooling margin monitors is considered to be essential, since all plant transients which could lead to ICC conditions will result in a loss of subcooled margin, regardless of the initiating event. Subcooling monitors are currently included in Technical Specifications. In addition, core exit thermocouples are used in certain accident sequences and are likewise retained. Review of the Oconee 3 PRA indicate that this instrument/variable is not utilized in any risk-significant event.

The NRC staff has recently acknowledged the above position within a safety evaluation report (SER) for TMI-1 dated December 13, 1988 as well as an April 19, 1985 SER for San Onofre. It is likely that similar NRC staff conclusions are contained in SERs for Turkey Point, Palo Verde, and Seabrook; however, due to the expedited nature of this submittal, Duke has not had the opportunity to review these documents. Consistent verbal NRC staff approval on May 11, 1989 Duke is providing Technical Specifications including shutdown requirements (Attachment 2).

GL 83-37 Item II.F.2
Core Exit Thermocouples

Core exit thermocouples are used as the key indication to identify superheated conditions following a loss of subcooling. Core exit thermocouples are also utilized as an input into the core subcooling margin determination which is used by operators during natural circulation conditions. Duke has previously proposed Technical Specification amendments which require shutdown in the event of core exit thermocouple inoperability. Consistent with verbal NRC staff approval on May 11, 1989 Duke is providing Technical Specifications including shutdown requirements (Attachment 2).

GL 83-37 Item II.F.2
Subcooling Margin Monitors

Subcooling margin monitors are utilized by the operator to determine when to trip reactor coolant pumps and to assure adequate core cooling during natural circulation. Existing Technical Specification 3.1.12 includes operability requirements for the subcooling monitors. Consistent with verbal NRC staff approval this Specification is being relocated to Specification 3.5.6 with revised allowable outage times (Attachment 2).

GL 83-37 Item III.D.3.4
Control Room Habitability

Theoretically, in-plant releases of chemical vapors or radioactive effluents call for use of the control room pressurization and filtering system to prevent in-leakage. However, no Design Basis Accident (DBA) is identified that could result in concentrations above NRC criteria, i.e., chlorine, hydrazine, gas decay tanks, etc. The commitment to pressurize following a Loss of Coolant Accident (LOCA) is based on potential iodine sources from containment leakage into the penetration room and into the control room bypassing filtration. However, consideration of thyroid doses from iodine has been deferred until resolution of the current source term reevaluation study by NRC Safety Evaluation dated November 24, 1986. Upon completion of the NRC reevaluation, Duke will review the results to determine appropriate actions, if any, to be taken. With the deferral of thyroid doses from iodine the effect of infiltration on the overall noble gas whole body dose in the control room will be negligible. Thus, in the calculation of post-LOCA operator doses, the concentration of radionuclides in the control room is assumed to be the same as the concentration in the supply intake. In summary, control room pressurization, filter train operability, supply air flow rate, and infiltration rates have no impact on the dose analysis in the absence of iodine considerations. Operability of this system is not part of the licensing basis for Oconee. This system was not designed with the intent of meeting GDC-19.

It is premature to include shutdown requirements for a system in which technical issues remain open. However, consistent with verbal NRC staff approval on May 11, 1989 Duke is providing Technical Specifications including shutdown requirements (Attachment 2).

The original proposal for Technical Specification 3.15 (dated January 6, 1988) included operability requirements for control room air handling unit (AHU) fans, due to the possible contribution for pressurization. Pressurization criteria is included within proposed Specification 3.15.2.6 (Attachment 2); any contribution from AHU fans is implicitly included within this specification.

Duke Power Company

Oconee Nuclear Station

Attachment 4

No Significant Hazards Consideration Evaluation

No Significant Hazards Consideration Evaluation

Duke has determined that the proposed amendment request passes no significant hazards as defined by NRC regulations in 10CFR50.92. 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;

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.

The commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (48FR14870) of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a surely administrative change to Technical Specifications. Example (ii) relates to a change which constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications. The changes proposed within this amendment request have been determined to be similar to either Example (i) or (ii).

The following evaluation measures aspects of this amendment request against the Part 50.92(c) requirements to demonstrate that all three standards are satisfied.

First Standard

(Amendment would not) involve a significant increase in the probability or consequences of an accident previously evaluated.

> The proposed amendments addressed in this submittal constitute either additional restrictions not presently included in the Technical Specifications or changes which are purely administrative in nature. Each accident analysis in the Oconee FSAR has been examined with respect to changes proposed in this amendment request. The probability of any Design Basis Accident (DBA) is not affected by this change, nor are the consequences of a DBA affected by this change, since additional requirements for operability and surveillance of accident monitoring instrumentation and control room habitability systems are not considered to be an initiator or contributor to any accident analysis addressed in the Oconee FSAR. As such, this change will not involve a significant increase in the probability or consequences of previously evaluated accidents.

Second Standard

(Amendment would not) create the possibility of a new or different kind of accident previously evaluated.

Changes provided within this amendment request constitute either additional restrictions not presently included in the Technical Specifications or changes which are purely administrative in nature. Consequently, this change will not create the possibility of a new or different kind of accident from any kind of accident previously evaluated.

Third Standard

(Amendment would not) involve a significant reduction in a margin of safety.

These changes constitute either additional restrictions not presently included in Technical Specifications or changes which are purely administrative in nature. Additional requirements for operability and surveillance of accident monitoring instrumentation and control room habitability systems will not impact any margins of safety. As such, there will be no reduction in any margin of safety.

Duke has concluded based on the above and the supporting Technical Justification in Attachment 3 that there is a No Significant Hazards Consideration involved in this amendment request.