Duke Power Company
Oconee Nuclear Station

Attachment 1

Proposed Technical Specification Revision

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Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic	NA	MO	NA	
2. Control Rod Drive Trip Breaker	NA	MO	NA	
3. Power Range Amplifier	ES(1)	NA	(1)	(1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	MO	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Intermediate Range	ES(1)	PS	NA	(1) When in service.
6. Source Range	ES(1)	PS	NA	(1) When in service.
7. Reactor Coolant Temperature	ES	MO	RF	
8. High Reactor Coolant Pressure	ES	MO	RF	
9. Low Reactor Coolant Pressure	ES	MO	RF	
**10. Flux-Reactor Coolant Flow Comparator	ES	MO	RF	
11. Reactor Coolant Pressure Temperature Comparator	ES	MO	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
12. Pump-Flux Comparator	ES	MO	RF	
13. High Reactor Building Pressure	DA	MO	RF	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems
**15. High Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
16. Low Pressure Injection Logic	NA	MO	NA	
**17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems
**19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
20. Reactor Building System Logic	NA	MO	NA	
**21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure	NA	MO	RF	
22. Pressurizer Temperature	ES	NA	RF	
**23. Control Rod Absolute Position	ES(1)	NA	RF(2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel.
**24. Control Rod Relative Position	ES(1)	NA	RF(2)	(1) Check with Absolute Position Indicator. (2) Calibrate rod misalignment channel.
25. Core Flood Tanks				
a. Pressure	ES	NA	RF	
b. Level	ES	NA	RF	
**26. Pressurizer Level	ES	NA	RF	
27. Letdown Storage Tank Level	DA	NA	RF	
28. Radiation Monitoring Systems	WE(1)	MO	QU	(1) Check functioning of self-checking feature on each detector.
29. High and Low Pressure Injection Systems Flow Channels	NA	NA	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
30. Borated Water Storage Tank Level Indicator	WE	NA	RF	
31. Boric Acid Mix Tank;				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	
32. Concentrated Boric Acid Storage Tank:				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	
33. Containment Temperature	NA	NA	RF	
34. Incore Neutron Detectors	MO(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout.
35. Emergency Plant Radiation Instruments	MO(1)	NA	RF	(1) Battery check.
36. Environmental Monitors	MO(1)	NA	RF	(1) Check functioning.
37. Reactor Manual Trip	NA	PS	NA	
**38. Reactor Building Emergency Sump Level	NA	NA	RF	
**39. Steam Generator Water Level	WE	NA	RF	
40. Turbine Overspeed Trip	NA	NA	RF	

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	

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

* This Technical Specification will become effective as follows:
Unit 1 - at the first convenient outage prior to or at the end of Oconee 1 Cycle 8 Refueling Outage
Unit 2 - end of Oconee 2 Cycle 6 Refueling Outage
Unit 3 - end of Oconee 3 Cycle 7 Refueling Outage
During the interim period, these discharge pressure switches will be tested during cold shutdown not to exceed once per month.

**The requirements of Specification 4.0.2 for surveillances performed on a refueling outage schedule are waived for Unit 1 until 11:59 July 15, 1983.

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
1	Pressurizer liquid sample line (Unit 1 only)	Note 1	Type C	Note 2, 7b
2	OTSG A Sample line	Note 1	Type C	Note 7b
3	Component cooling inlet line	Note 1	Type C	Note 3, 7d
4	OTSG B drain line	Note 1	None required	Note 7b
5	RB normal sump drain line	Note 10	Type C	Note 7a, 7b, 9
6	Letdown line	Note 1	Type C	Note 2, 7b
7	RC Pump seal return line	Note 1	Type C	Note 3, 7b, 9
8	Loop A nozzle warming line	Not Vented	None required	Note 5, 7d
9	RCS normal makeup line and HP injection 'A' loop	Not Vented	None required	Note 5
10	RC Pump seal injection	Not Vented	Type C	Note 5, 7d, 9, 12

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
36 37	RB emergency sump recirculation line	Not Vented	None required	Note 5
38	Quench tank cooler inlet line	Note 1	Type C	Note 2, 7d, 12
39	HP Nitrogen supply	Note 1	None required	Note 3 (manual valves)
(Unit 2, 3) Only	CFT Vent line	Note 1	None required	Note 3 (manual valves)
40	RB emergency sump drain line	Note 1	None required	Note
41	Instrument air supply & ILRT verification line	Note 1	None required	Note 3 (manual valves)
42	SPARE	Not in Use		
43	OTSG A drain line	Note 1	None required	Note 7b
44	Component cooling to control rod drive inlet line	Note 1	Type C	Note 3, 7d
45	ILRT instrument line	Not Vented	Type C	Note 3, 7a
46	Reactor head-wash filtered water inlet	Note 1	Type C	Note 3 (manual valves)

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
47 (Unit 1 only)	Demineralized water supply to RC pump seal vents	Note 1	Type C	Note 3, 7d
48	Breathing air inlet	Note 1	None required	Note 3 (manual valves)
49 (Unit 1 only)	LP Nitrogen supply	Note 1	None required	Note 3 (manual valves)
50	OTSG A Emergency FDW line	Not Vented	None required	Note 5
51	ILRT Pressurization line	Note 1	None required	Note 6a, 7a
52	HP Injection to 'B' loop	Not Vented	None required	Note 5
53 (All)	HP Nitrogen supply to 'A' core flood tank	Note 1	None required	Note 3 (manual valves)
(Unit 2, 3)	LP Nitrogen supply	Note 2	None required	Note 3 (manual valves)
54	Component cooling outlet line	Note 1	Type C	Note 3, 7b, 9(8)
55	Demineralized water supply	Note 1	Type C	(Unit 1) Note 3, (manual valves), 12 (Unit 2,3) Note 3, 9 (manual valves)
56	Spent fuel canal fill and drain	Note 1	None required	Note 3 (manual valve)
57 (Unit 1 only)	DHR return line	Not Vented	None required	Note 4

TABLE 4.4-1
LIST OF PENETRATIONS WITH 10CFR50,
APPENDIX J TEST REQUIREMENTS

PENETRATION NUMBER	SYSTEM	TYPE A TEST SYSTEM CONDITION	LOCAL LEAK TEST	REMARKS
58 (All)	OTSG B sample line	Note 1	Type C	Note 7b
(Unit 2, 3)	Pressurizer sample line	Note 1	Type C	Note 2, 7b
59	CF tank sample line	Note 1	None required	Note 2
60	RB sample line (outlet)	Note 1	Type C	Note 2, 7b, 9
61	RB sample line (inlet)	Note 1	Type C	Note 3, 7b, 9
62 (Units 2, 3 only)	DHR return line	Not vented	None required	Note 4
	Personnel hatch	Vented	Type B	Note 6b
	Emergency hatch	Vented	Type B	Note 6b
	Equipment hatch	Vented	Type B	Note 6c
	Electrical penetration	Vented	Type B	Note 6a, 12

TABLE 4.4-1
NOTES (continued)

c. Isolation valves are required to operate intermittently under post accident conditions.

d. Check valves used for containment isolation.

NOTE 8 DELETED

NOTE 9 Reverse direction test of inside containment isolation valve authorized. Leakage results are conservative.

NOTE 10 System is submerged during post-accident conditions and performance of Type A test. System will be drained to the extent possible.

NOTE 11 Type B test performed on the blind flanges inside the Reactor Building. The tube drain valves and valves outside the containment are not tested.

NOTE 12 The requirements of Section 4.4.1.2.2 are waived for this penetration in Unit 1 until 11:59 p.m. July 15, 1983.

4.6 EMERGENCY POWER PERIODIC TESTING

Applicability

Applies to the periodic testing surveillance of the emergency power sources.

Objective

To verify that the emergency power sources and equipment will respond promptly and properly when required.

Specification

4.6.1 Monthly, a test of the Keowee Hydro units shall be performed to verify proper operation of these emergency power sources and associated equipment. This test shall assure that:

a. Each hydro unit can be automatically started from the Unit 1 and 2 control room.

b. Each hydro unit can be synchronized through the 230 kV overhead circuit to the startup transformers.

c. Each hydro unit can energize the 13.8 kV underground feeder.

d. The 4160 volt startup transformer main feeder bus breakers and standby bus breaker shall be exercised.

4.6.2 a. Annually, the Keowee Hydro units will be started using the emergency start circuits in each control room to verify that each hydro unit and associated equipment is available to carry load within 25 seconds of a simulated requirement for engineered safety features.

b. Promptly following the above annual test, each hydro unit will be loaded to at least the combined load of the auxiliaries actuated by ESG signal in one unit and the auxiliaries of the other two units in hot shutdown by synchronizing the hydro unit to the offsite power system and assuming the load at the maximum practical rate.

4.6.3 Monthly, the Keowee Underground Feeder Breaker Interlock shall be verified to be operable.

*4.6.4 During each refueling outage, a simulated emergency transfer of the 4160 volt main feeder buses to the startup transformer (i.e., CT1, CT2 or CT3) and to the 4160 volt standby buses shall be made to verify proper operation.

4.6.5 Quarterly, the External Grid Trouble Protection System logic shall be tested to demonstrate its ability to provide an isolated power path between Keowee and Oconee.

*The requirement of Specification 4.0.2 for surveillance performed on a refueling outage schedule is waived for Section 4.6.4 on Unit 1 until 11:59 p.m. July 15, 1983.

4.18 SNUBBERS

Applicability

Applies to hydraulic and mechanical snubbers used to protect the Reactor Coolant System and other safety-related systems.

Objective

To verify that the required hydraulic and mechanical snubbers are operable.

Specification

*4.18.1 Each snubber associated with the Reactor Coolant System and other safety-related systems, as specified in the appropriate Station Procedure shall be visually inspected. Visual inspections shall verify:

(1) that there are no visible indications of damage or impaired OPERABILITY,

(2) attachments to the foundation or supporting structure are secure, and

(3) in those locations where mechanical snubber movement can be manually induced, the snubbers shall be inspected as follows:

(a) At each refueling, the inaccesible snubbers shall be inspected near the beginning and the end of the outage.

(b) In the event of a severe dynamic event, snubbers in that system which experienced the event shall be inspected during the refueling outage to assure that the snubbers have freedom of movement and are not frozen up. The inspection shall consist of verifying freedom of motion using one of the following: (i) Manually induced snubber movement, (ii) evaluation of in place snubber piston setting; (iii) stroking the mechanical snubber through its full range of travel. If one or more mechanical snubbers are found to be frozen up during this inspection, those snubbers shall be replaced (or overhauled) before returning to power. Re-inspection shall subsequently be performed according to the schedule listed below.

Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.18.4. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be tested

*The inspection period requirements of Section 4.18.1 are waived for the inaccessible mechanical snubbers on Unit 1 until 11:59 p.m. July 15, 1983.

Attachment 2

Page Number	Date Procedure Last Performed (II.A)	Date Due (II.B)
C-1	July 12, 1981	May 28, 1983
D-1	August 21, 1981	July 6, 1983
E-1	June 26, 1981	May 11, 1983
F-1	June 29, 1981	May 14, 1983
G-1	June 29, 1981	May 14, 1983
H-1	June 29, 1981	May 14, 1983
I-1	July 6, 1981	May 22, 1983
J-1	July 6, 1981	May 22, 1983
K-1	July 9, 1981	May 25, 1983
L-1	July 1, 1981	May 17, 1983
M-1	July 6, 1981	May 22, 1983
N-1	July 9, 1981	May 25, 1983
O-1	July 17, 1981	June 2, 1983
P-1	July 17, 1981	June 2, 1983
Q-1	July 23, 1981	June 8, 1983
R-1	July 21, 1981	June 6, 1983
S-1	July 22, 1981	June 7, 1983
T-1	July 23, 1981	June 8, 1983
U-1	August 5, 1981	June 20, 1983
X-1	July 21, 1981	June 6, 1983

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

1) Section 4.4.1.2.2; Frequency of Test for the Local Leak Rate Testing.

2) In accordance with the criteria specified in Appendix J of 10CFR50.

(B) Structure/System/Component/Parameter Effected:

Electrical Penetration O-ring Seal

(C) Test/Calibration/Inspection Procedure name:

Electrical Penetration O-Ring Seal Leak Test

(D) Test/Calibration/Inspection Procedure Number:

PT/O/A/150/5

(E) Test/Calibration/Inspection Function:

To determine the leak rate through the double o-ring seal between electrical penetrations and Reactor Building flanges.

II. Bases for Requesting Relief

(A) Date procedure last performed:

The electrical penetration o-ring seal leak test for Unit 1 began on June 30, 1981 and was completed by July 16, 1981.

(B) Date Due:

June 30, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 30 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

III. Justification for Relief

Five Unit 1 pipe supports (04A-0-478A-NPS-H35, H35A, H37, H59 and NPS-04A-0-479-H41) that contain mechanical snubbers were inspected during Unit 1's 1981 refueling and 10 year inservice inspection outage. An evaluation for the effects on system operability if these snubbers were assumed to be inoperable was performed.

Based on this evaluation, it was determined that system operability is assured if supports H35, H35A and H37 were to be inoperable, due to the low stress conditions that exist in the piping and the reserve capacities that adjacent pipe supports possess. Support H59 is required to assure system operability; however, this support had a new snubber installed near the end of the Unit 1 outage per the requirements of IEB 79-14. Follow-up discussions confirmed that this support was installed and stroke tested in October, 1981. Thus, this supports next required inspection would be August, 1983. Support H41 was determined to be located on non-safety piping. This support was installed as part of SMR 31S. Subsequent review of the adjacent safety related piping for IEB 79-14 and NSM 1012-3108 did not require any seismic supports on this portion of the non-safety piping. H41 has no effect on the operability of either the safety or non-safety piping and can thus be left in place or removed.

DUKE POWER COMPANY

OCONEE NUCLEAR STATION
REQUEST FOR RELIEF FROM
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS

I. Surveillance Requirements for which Relief is Requested

(A) Technical Specification Effected:

Section 4.1.1 - Frequency and type of surveillance required per Table 4.1-1 Item 38.

(B) Structure/System/Component/Parameter Effected:

Emergency Sump Level Indication System

(C) Test/Calibration/Inspection Procedure Name:

Low Pressure Injection System Reactor Building Emergency Sump Level Instrument Calibration

(D) Test/Calibration/Inspection Procedure Number:

IP/O/A/203/1E

(E) Test/Calibration/Inspection Function:

To calibrate the Reactor Building Emergency Sump Level Instrumentation System.

II. Bases for Requesting Relief

(A) Date procedure last performed:

August 10, 1981

(B) Date Due:

June 25, 1983

(C) Plant Status:

It is anticipated that Unit 1 will have approximately 15 EFPD left in its cycle, operating at 100% full power, with no planned outages until the refueling outage beginning July 3, 1983.

(D) Required Surveillance Frequency:

Technical Specification 4.0.2 requires that the maximum allowable interval between surveillances be 22 months, 15 days.