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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 94
License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Arkansas Power & Light Company (the licensee) dated March 16, 1984 as supplemented August 22, 1984 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 94 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 31, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 94

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
--	16b
17	17
42a	42a
45d	45d
45f	45f
71	71
72d	72d
--	73b
127	127
146	146

3.1.1.7 Reactor Coolant System Vents

At least one reactor coolant system vent path consisting of at least two valves in series shall be operable at each of the following locations whenever the Reactor Coolant average temperature is above 280°F.

1. Reactor Vessel head

2. Pressurizer steam space

3. Reactor coolant system Hot Leg high points (2 locations)

A. With one of the above vent paths inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed; restore the inoperable vent path to operable status within 30 days, or be in hot standby within 6 hours and in cold shutdown within the following 30 hours.

B. With two or more of the above vent paths inoperable, maintain the inoperable vent paths closed and restore at least two vent paths to operable status within 72 hours or be in hot standby within 6 hours and in cold shutdown within the following 30 hours.

BASES:

The plant is designed to operate with both reactor coolant loops and at least one reactor coolant pump per loop in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients.(1)

Whenever the reactor coolant average temperature is above 280°F, single failure considerations require that two loops be operable.

The decay heat removal system suction piping is designed for 300°F, thus, the system can remove decay heat when the reactor coolant system is below this temperature.(2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat.(4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident.(5) The pressurizer code safety valve lift set point shall be set at 2,500 psig ±1 percent allowance for error and each valve shall be capable of relieving 300,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internal vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

The reactor coolant vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head, the reactor coolant system high points, and the pressurizer steam space ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are covered in Section 4.0 for the class 2 valves and Table 4.1-2 for the vent paths. These are consistent with ASME Section XI and Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," 11/80.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7
(2) FSAR, Section 4.2.5.1 and 9.5.2.3
(3) FSAR, Section 4.2.5.4
(4) FSAR, Section 4.3.10.4 and 4.2.4
(5) FSAR, Section 4.3.7

3.5.1.7 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. The relief valve setting for the DHR system shall be equal to or less than 450 psig.

3.5.1.8 The degraded voltage monitoring relay settings shall be as follows:

a. The 4.16 KV emergency bus undervoltage relay setpoints shall be > 3115 VAC but < 3177 VAC.

b. The 460 V emergency bus undervoltage relay setpoints shall be > 423 VAC but < 431 VAC with a time delay setpoint of 8 seconds ±1 second.

3.5.1.9 The following Reactor Trip circuitry shall be operable as indicated:

1. Reactor trip upon loss of Main Feedwater shall be operable (as determined by Specification 4.1.a, items 2 and 36 of Table 4.1-2) at greater than 5% reactor power. (May be bypassed up to 10% reactor power.)

2. Reactor trip upon Turbine Trip shall be operable (as determined by Specification 4.1.a, items 2 and 42) at greater than 5% reactor power. (May be bypassed up to 20% reactor power.)

3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition.

3.5.1.10 The control room ventilation chlorine detection system instrumentation shall be operable and capable of actuating control room isolation and filtration systems, with alarm/trip setpoints adjusted to actuate at a chlorine concentration of ≤ 5ppm.

3.5.1.11 For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any one time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed.

3.5.1.12 The Containment High Range Radiation Monitoring instrumentation shall be operable with a minimum measurement range from 1 to 10^7 R/hr.

TABLE 3.5.1-1 (Cont'd)

OTHER SAFETY RELATED SYSTEMS (Cont'd)

	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
2. Pressurizer level channels	2	N/A	2	1	Note 10
3. Emergency Feedwater Flow channels	2/S.G.	N/A	1	0	Note 10
4. RCS subcooling margin monitors	2	N/A	1	0	Note 10
5. Electromatic relief valve flow monitor	2	N/A	1	0	Note 11
6. Electromatic relief block valve position indicator	1	N/A	1	0	Note 12
7. Pressurizer code safety valve flow monitors	2/valve	N/A	1/valve	0	Note 10
8. Degraded Voltage Monitoring					
a. 4.16KV Emergency Bus Undervoltage	2/Bus	1/Bus	2/Bus	0	Note 14
b. 460V Emergency Bus Undervoltage	*1/Bus	1/Bus	1/Bus	0	Notes 13, 14
9. Chlorine Detection Systems	2	1	2	0	Notes 17, 18
10. Containment High Range Radiation Monitoring	2	N/A	2	0	Note 19
11. Containment Pressure - High Range	2	N/A	2	0	Note 20
12. Containment Water Level - Wide Range	2	N/A	2	0	Note 20

*Two undervoltage relays per bus are used with a coincident trip logic (2-out-of-2)

TABLE 3.5.1-1 (Cont'd)

12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours.

13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies.

14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

15. This trip function may be bypassed at up to 10% reactor power.

16. This trip function may be bypassed at up to 20% reactor power.

17. With no channel operable, within 1 hour restore the inoperable channels to operable status, or initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

19. This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig.

20. With one channel inoperable, (1) either restore the inoperable channel to operable status within 7 days, or (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.4 within 30 days following the event, outlining the action taken, the cause of the inoperab lity, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

21. With one channel inoperable, restore the inoperable channel to operable status within 30 days or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage. If both channels are inoperable, restore the inoperable channels within 30 days or be in hot shutdown within 12 hours.

Table 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks
20.	Reactor Building Spray System System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valve, and chem. add. valve logic channels.
21.	Reactor Building Spray System Analog Channels				
	a. Reactor Building Pressure Channels	NA	M	R	
22.	Pressurizer Temperature Channels	S	NA	R	
23.	Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.
24.	Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator
25.	Core Flooding Tanks				
	a. Pressure Channels	S	NA	R	
	b. Level Channels	S	NA	R	
26.	Pressurizer Level Channels	S	NA	R	
27.	Makeup Tank Level Channels	D	NA	R	
28.	Radiation Monitoring Systems other than containment high range monitors (item 55)	W	M(1)	Q(2)	(1) Check functioning of self-checking feature on each detector. (2) R for those detectors inaccessible during normal operation.
29.	High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	

TABLE 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
d. SG A high range level high-high	S	M	R	
e. SG B high range level high-high	S	M	R	
57. Containment High Range Radiation Monitors	D	M	R	
58. Containment Pressure-High	M	NA	R	
59. Containment Water Level-Wide Range	M	NA	R	

Note:

S-Each Shift
W-Weekly
M-Monthly
D-Daily

T/W-Twice per Week
Q-Quarterly
P-Prior to each startup if not done previous week
B/M-Every 2 Months

R-Once every 18 months
PC-Prior to going Critical if not done within previous 31 days
NA-Not applicable

TABLE 4.1-2 (Continued)

Minimum Equipment Test Frequency

	Item	Test	Frequency
16.	RCS Vent Paths	Demonstrate operability by flow verification	At least once per 18 months during cold shutdown

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a. The facility shall be placed in at least hot shutdown within one hour.

b. The Nuclear Regulatory Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.12.3.1.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. New and spent fuel storage

h. Offsite Dose Calculation Manual and Process Control Program implementation at the site.

i. Post accident sampling (includes sampling of reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and the containment atmosphere).

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the ANO General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PSC and approved by the ANO General Manager within 14 days of implementation.

6.12.4 Unique Reporting Requirements

Unique reports cover inspections, tests, maintenance, and special reports that are appropriate to assure safe operation of the plant. The frequency and content of these reports are determined on an individual case basis and designated in these Technical Specifications. Unique reports shall be submitted in writing to the appropriate Regional Office within 90 days of the completion of the tests, inspections and maintenance unless indicated otherwise within the referenced specification.

The subjects of unique reports shall include:

(a) Tendon surveillance. (Specification 4.4.2)

(b) Inoperable containment radiation monitors (Specification 3.5.1.12)

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 63
License No. NPF-6

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Arkansas Power & Light Company (the licensee) dated March 16, 1984 as supplemented August 22, 1984 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-6 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, except where otherwise stated in specific license conditions.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 31, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 63

FACILTIY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove Pages	Insert Pages
VI	VI
XII	XII
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
3/4 3-40	3/4 3-40
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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

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BASES

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TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area Monitor	1	*	$\leq 1.5 \times 10^{-2}$ R/hr	$10^{-4} - 10^{1}$ R/hr	13
b. Containment High Range	2	1, 2, 3 & 4	Not Applicable	$1 - 10^{7}$ R/hr	18
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity					
a) Purge & Exhaust Isolation	1	ALL MODES	$\leq$ 2 x background	$10 - 10^{6}$ cpm	16
b) RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 - 10^{6}$ cpm	14
ii. Particulate Activity					
a) RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 - 10^{6}$ cpm	14
b. Control Room Ventilation Intake Duct Monitor	1	ALL MODES	$\leq$ 2 x background	$10 - 10^{6}$ cpm	17

* With fuel in the spent fuel pool or building

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

ACTION 17 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

ACTION 18 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, (1) either restore the inoperable channel to OPERABLE status within 7 days or (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Spent Fuel Pool Area Monitor	S	R	M	*
b. Containment High Range	S	R****	M	1, 2, 3, & 4
2. PROCESS MONITORS				
a. Containment				
i. Gaseous Activity				
a) Purge & Exhaust Isolation	**	R	***	ALL MODES
b) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Particulate Activity				
a) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
b. Control Room Ventilation Intake Duct Monitor	S	R	M	ALL MODES

* With fuel in the spent fuel pool or building.

** Within 8 hours prior to initiating containment purge operations and at least once per 12 hours during containment purge operations.

*** Within 31 days prior to initiating containment purge operations and at least once per 31 days during containment purge operations.

**** Acceptable criteria for calibration are provided in Table II.F.1-3 of NUREG-0737.

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detectors with at least one incore detector in each quadrant at each level, and

b. At least 75% of all incore detector locations, and

c. Sufficient operable incore detectors to perform at least six tilt estimates with at least one tilt estimate at each of three levels.

An OPERABLE incore detector location shall consist of a fuel assembly containing either a fixed detector string with a minimum of three OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

A tilt estimate can be made from two sets of symmetric pairs of incore detectors. Two sets of symmetric pairs of incore detectors are formed by two pairs of diagonally opposite symmetric incore detectors, one incore detector per quadrant.

APPLICABILITY: When the incore detection system is used for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin.

ACTION:

With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each post-accident monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3-10.

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE
1. Containment Pressure (Normal Design Range)	2
2. Containment Pressure (High Range)*	2
3. Pressurizer Pressure	2
4. Pressurizer Water Level	2
5. Steam Generator Pressure	2/steam generator
6. Steam Generator Water Level	2/steam generator
7. Refueling Water Tank Water Level	2
8. Containment Water Level - Wide Range*	2
9. Emergency Feedwater Flow Rate	1/steam generator
10. Reactor Coolant System Subcooling Margin Monitor	1
11. Pressurizer Safety Valve Acoustic Position Indication	1
12. Pressurizer Safety Valve Tail Pipe Temperature	1

*If only one channel is inoperable and containment entry is required to restore the inoperable channel, the channel need not be restored until the following refueling outage.

TABLE 4.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure (Normal Design Range)	M	R
2. Containment Pressure (High Range)	M	R
3. Pressurizer Pressure	M	R
4. Pressurizer Water Level	M	R
5. Steam Generator Pressure	M	R
6. Steam Generator Water Level	M	R
7. Refueling Water Tank Water Level	M	R
8. Containment Water Level - Wide Range	M	R
9. Emergency Feedwater Flow Rate	M	R
10. Reactor Coolant System Subcooling Margin Monitor	M	R
11. Pressurizer Safety Valve Acoustic Position Indication	M	R
12. Pressurizer Safety Valve Tail Pipe Temperature	M	R

INSTRUMENTATION

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.7 Two independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of $\leq$ 5 ppm, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one chlorine detection system inoperable, restore the inoperable detection system to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

b. With no chlorine detection system OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.7 Each chlorine detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor coolant system vent path consisting of at least two valves in series shall be OPERABLE at each of the following locations:

1. Reactor Vessel Head

2. Pressurizer Steam Space (RCS High Point Vents)

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With less than one vent path from each of the locations OPERABLE, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path(s) is maintained closed; restore the inoperable vent path to OPERABLE status within 30 days or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both vent paths 1 and 2 above inoperable, restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor coolant system vent path shall be demonstrated OPERABLE at least once per 18 months by verifying flow through the reactor coolant vent system vent paths during COLD SHUTDOWN.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a(g)(6)(i).

ASME Code Class 1 components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition and Addenda through Summer 1973. ASME Code Class 2 and 3 components were designed to provide access to permit inservice inspecitons in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition and Addenda through Winter 1972.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

The reactor coolant vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one vent path from the reactor vessel head and the reactor coolant system high point ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are consistent with ASME Section XI for Class 2 valves (see 3/4.4.10 above) and Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Vice President, Nuclear Operations and to the SRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Vice-President, Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants

NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the Plant Safety Committee.

h. New and spent fuel storage.

i. ODCM and PCP implementation.

j. Postaccident sampling (includes sampling of reactor coolant radioactive iodines and particulates in plant gaseous effluent, and the containment atmosphere).

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the ANO General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PSC and approved by the ANO General Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Administrator of the Regional Office unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Administrator of the Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

e. Inoperable Fire Detection Instrumentation, Specification 3.3.3.8.

f. Inoperable Fire Suppression Systems, Specifications 3.7.10.1 and 3.7.10.2.

g. Primary Coolant Specific Activity, Specification 3.4.8.

h. Radioactive Effluents, Specifications 3.11.1.1, 3.11.1.2, 3.11.1.3, 3.11.2.1, 3.11.2.2, 3.11.2.3, and 3.11.2.4 and 3.11.2.5.

   This report shall include the following:

   1) Description of the occurrence.
   2) Identify the cause(s) for exceeding the limit(s).
   3) Explain corrective action(s) taken to mitigate occurrence.
   4) Define action(s) taken to prevent recurrence.
   5) Summary of consequence(s) of occurrence.

i. Inoperable Containment Radiation Monitors, Specification 3.3.3.1.

SEMIANNUAL RADIOLOGICAL EFFLUENT RELEASE REPORT*

6.9.3 Routine radioactive effluent release reports covering the operating of the unit during the previous 6 months of operations shall be submitted within 60 days after January 1 and July 1 of each year.

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*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste system, the submittal shall specify the releases of radioactive material from each unit.

6.9.3.1 The radioactive effluent release report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste release from the unit. The data will be summarized on a quarterly basis following the format of Regulatory Guide 1.21, Revision 0, Appendix A.

6.9.3.2 Any changes in the OFFSITE DOSE CALCULATION MANUAL and PCP shall be included in the semiannual report for the period in which the change(s) was made effective.

6.9.3.3 The radioactive effluent release reports shall include the following information for all unplanned releases to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents:

1. Description of the occurrence.
2. Identify the cause(s) for exceeding the limit(s).
3. Explain corrective actions taken to mitigate occurrence.
4. Define action(s) taken to prevent recurrence.
5. Summary of consequence(s) of occurrence.

6.9.3.4 The first report filed each year shall contain:

1. A summary of the hourly meteorological data collected over the previous calendar year. In lieu of including this summary in the report, the data may be retained by the licensee for NRC review and noted as such in the report.
2. A summary of radiation doses due to radiological effluent during the previous calendar year calculated in accordance with the methodology specified in the OFFSITE DOSE CALCULATION MANUAL.
3. The radiation dose to members of the public due to their activities inside the site boundary. This calculated dose shall include only those dose contributions directly attributed to operation of the unit and shall be compared to the limits specified in 40 CFR 190.

6.9.3.5 The first report filed each year shall contain description of licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid) during the previous calendar year.*

*This information may be included in the annual FSAR update in lieu of inclusion in this report.