Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

July 13, 1998

SUBJECT: ISSUANCE OF AMENDMENT NOS. 192 AND 191 TO FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6 - ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 (TAC NOS. M96893 AND M96894)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment Nos. 192 and 191 to Facility Operating License Nos. DPR-51 and NPF-6 for the Arkansas Nuclear One, Units 1 and 2 (ANO-1&2). These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated October 2, 1996.

The amendments revise the ANO-1&2 TSs by relocating selected TS requirements related to instrumentation from the TS to the Updated Final Safety Analysis Report. The Nuclear Regulatory Commission provided guidance to all holders of operating licenses or construction permits for nuclear power reactors on the proposed TS changes in Generic Letter 95-10, "Relocation of Selected Technical Specifications Requirements Related to Instrumentation," dated December 15, 1995.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
ORIGINAL SIGNED BY:
William Reckley, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-313 and 50-368

Enclosures:
1. Amendment No. 192 to DPR-51
2. Amendment No. 191 to NPF-6
3. Safety Evaluation

cc w/encls: See next page

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Document Name: ARM96893.AMD w/change to SE p.1

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 13, 1998

Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ISSUANCE OF AMENDMENT NOS. 192 AND 191 TO FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6 - ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 (TAC NOS. M96893 AND M96894)

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Sincerely,

Will Reckley

William Reckley, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-313 and 50-368

Enclosures:
1. Amendment No. 192 to DPR-51
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3. Safety Evaluation

cc w/encls: See next page

Mr. C. Randy Hutchinson
Entergy Operations, Inc.

Arkansas Nuclear One, Units 1 & 2

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 192
License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated October 2, 1996, 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 license 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.

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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 Appendix A, as revised through Amendment No. 192, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented at the facility within 30 days (including issuance of Technical Requirements Manual for use by licensee personnel). In addition, the licensee shall include the relocated information in the Updated Final Safety Analysis Report submitted to the NRC, pursuant to 10 CFR 50.71(e), as was described in the licensee's application dated October 2, 1996, and evaluated in the staff's safety evaluation dated July 13, 1998.

FOR THE NUCLEAR REGULATORY COMMISSION

William Reckley

William Reckley, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 13, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 192

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

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

REMOVE PAGES	INSERT PAGES
42a	42a
42b	42b
43b	43b
43c	43c
45d	45d
45d1	45d1
45d2	-
45g	45g
72	72
72a	72a
72b	72b
72b1	72b1

3.5.1.7 The Dec Heat Removal System isolation ve 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 and item 35 of Table 4.1-1) 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 and item 41 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 45% 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 Deleted

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.

3.5.1.13 Deleted

3.5.1.14 The Main Steam Line Radiation Monitoring Instrumentation shall be operable with a minimum measurement range from 10^{-1} to 10^4 mR/hr, whenever the reactor is above the cold shutdown condition.

3.5.1.15 Initiate functions of the EFIC system which are bypassed at cold shutdown conditions shall have the following minimum operability conditions:

a. "low steam generator pressure" initiate shall be operable when the main steam pressure exceeds 750 psig.

b. "loss of 4 RC pumps" initiate shall be operable when neutron flux exceeds 10% power.

c. "main feedwater pumps tripped" initiate shall be operable when neutron flux exceeds 10% power.

3.5.1.16 The automatic steam generator isolation system within EFIC shall be operable when main steam pressure is greater than 750 psig.

Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state, on-line repairs to the failed device, when practical, will be made and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and, in many cases, make on-line repairs.

The Degraded Voltage Monitoring relay settings are based on the short term starting voltage protection as well as long term running voltage protection. The 4.16 KV undervoltage relay setpoints are based on the allowable starting voltage plus maximum system voltage drops to the motor terminals, which allows approximately 78% of motor rated voltage at the motor terminals. The 460V undervoltage relay setpoint is based on long term motor voltage requirements plus the maximum feeder voltage drop allowance resulting in a 92% setting of motor rated voltage.

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The subcooled margin monitors (SMM), and core-exit thermocouples (CET), Reactor Vessel Level Monitoring System (RVLMS) and Hot Leg Level Measurement System (HLLMS) are a result of the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737. The function of the ICC instrumentation is to increase the ability of the plant operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37 and are not required by the accident analysis, nor to bring the plant to cold shutdown conditions. The Reactor Vessel Level Monitor is provided as a means of indicating level in the reactor vessel during accident conditions. The channel operability of the RVLMS is defined as a minimum of three sensors in the upper plenum region and two sensors in the dome region operable. When Reactor Coolant Pumps are running, all except the dome sensors are interlocked to read "invalid" due to flow induced variables that may offset the sensor outputs. The channel operability of the HLLMS is defined as a minimum of one wide range and any two of the narrow range transmitters in the same channel operable. If the equipment is inaccessible due to health and industrial safety concerns (for example, high radiation area, low oxygen content of the containment atmosphere) or due to physical location of the fault (for example, probe failure in the reactor vessel), then operation may continue until the next scheduled refueling outage and a report filed.

To support loss of main feedwater analyses, steam line/feedwater line break analyses, SBLOCA analyses, and NUREG-0737 requirements, the EFIC system is designed to automatically initiate EFW when:

1. all four RC pumps are tripped

2. both main feedwater pumps are tripped

3. the level of either steam generator is low

4. either steam generator pressure is low

5. ESAS ECCS actuation (high RB pressure or low RCS pressure)

The EFIC system is also designed to isolate the affected steam generator on a steam line/feedwater line break and supply EFW to the intact generator according to the following logic:

- If both SG's are above 600 psig, supply EFW to both SG's.

- If one SG is below 600 psig, supply EFW to the other SG.

- If both SG's are below 600 psig, but the pressure difference between the two SG's exceeds 100 psig, supply EFW only to the SG with the higher pressure.

- If both SG's are below 600 psig and the pressure difference is less than 100 psig, supply EFW to both SG's.

At cold shutdown conditions all EFIC initiate and isolate functions are bypassed except low steam generator level initiate. The bypassed functions will be automatically reset at the values or plant conditions identified in Specification 3.5.1.15. "Loss of 4 RC pumps" initiate and "low steam generator pressure" initiate are the only shutdown bypasses to be manually initiated during cooldown. If reset is not done manually, they will automatically reset. Main feedwater pump trip bypass is automatically removed above 10% power.

REFERENCE

FSAR, Section 7.1

Table 3.5.1-1 (cont'd)

OTHER SAFETY RELATED SYSTEMS
(Cont'd)

Functional Unit	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. 4.16 KV Emergency Bus Undervoltage	2/Bus	1/Bus	2/Bus	0	Note 14
b. 460 V Emergency Bus Undervoltage	*1/Bus	1/Bus	1/Bus	0	Notes 13, 14
9. Deleted					
10. Containment High Range Radiation Monitoring	2	N/A	2	0	Note 20
11. Containment Pressure - High Range	2	N/A	2	0	Note 21
12. Containment Water Level - Wide Range	2	N/A	2	0	Note 21

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

Table 3.5.1-1 (cont'd)

OTHER SAFETY RELATED SYSTEMS
(Cont'd)

Functional Unit	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
13. In core Thermocouples (core-exit thermocouples)	6/core quadrant	N/A	2/core quadrant	0	Note 22
14. Deleted					
15. Reactor Vessel Level Monitoring System	2	N/A	2	0	Note 28, 29
16. Hot Leg Level Measurement System (HLLMS)	2	N/A	2	0	Note 28, 29
17. Main Steam Line Radiation Monitors	1/steam line	N/A	1/steam line	0	Note 30

23. With the number of operable Electronic (SCR) Trip relays one less than the total number of Electronic (SCR) Trip relays in a channel, restore the inoperable Electronic (SCR) Trip relay to operable status in 48 hours or place the SCRs associated with the inoperable Electronic (SCR) Trip relay in trip in the next hour. With two or more Electronic (SCR) Trip relays inoperable, place all Electronic (SCR) Trip relays associated with that channel in trip in the next hour. This requirement does not apply to the Electronic Trip channels associated with Group 8 Regulating Power Supply.

24. With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

    a. Within 1 hour:

        1. Place the inoperable channel in the tripped condition, or

        2. Remove power supplied to the control rod trip device associated with the inoperable channel.

    b. One additional channel may be bypassed for up to 4 hours for surveillance testing and the operable channel above may be bypassed for up to 30 minutes in any 24-hour period when necessary to test the trip breaker associated with the logic of the channel being tested. The inoperable channel above shall not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.

25. With one of the Control Rod Drive Trip Breaker diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

26. Interrupts motor power to the Safety Groups of control rods only.

27. Deleted

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	
30. Decay heat removal system isolation valve automatic closure and interlock system	S(1)(2)	M(1)(3)	R	(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) At least once every refueling shutdown, with Reactor Coolant System Pressure greater than or equal to 200 psig, but less than 300 psig, verify automatic isolation of the decay heat removal system from the Reactor Coolant System on high Reactor Coolant System pressure.
31. Deleted				
32. Diesel generator protective relaying starting interlocks and circuitry	M	Q	NA	
33. Off-site power undervoltage and protective relaying interlocks and circuitry	W	R(1)	R(1)	(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2.
34. Borated water storage tank level indicator	W	NA	R	
35. Reactor trip upon loss of main feedwater circuitry	M	PC	R	

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
36. Boric Acid Addition Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
37. Degraded Voltage Monitoring	W	R	R	
38. Sodium Hydroxide Tank Level Indicator	NA	NA	R	
39. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning
40. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check
41. Reactor Trip Upon Turbine Trip Circuitry	M	PC	R	
42. Deleted				

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
43. ESAS Manual Trip Functions				
a. Switches & Logic	NA	R	NA	
b. Logic	NA	M	NA	
44. Reactor Manual Trip	NA	P	NA	
45. Reactor Building Sump Level	NA	NA	R	
46. EFW Flow Indication	M	NA	R	

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
47. RCS Subcooling Margin Monitor	D	NA	R	
48. Electromatic Relief Valve Flow Monitor	D	NA	R	
49. Electromatic Relief Block Valve Position Indicator	D	NA	R	
50. Pressurizer Safety Valve Flow Monitor	D	NA	R	
51. Pressurizer Water Level Indicator	D	NA	R	
52. Deleted				
53. EFW Initiation				
a. Manual	NA	M	NA	
b. SG Low Level, SGA or B	S	M	R	
c. Low Pressure SGA or B	S	M	R	
d. Loss of both MFW Pumps and PWR > 10%	S	M	R	

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 191
License No. NPF-6

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated October 2, 1996, 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 license 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) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 191, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented at the facility within 30 days (including issuance of Technical Requirements Manual for use by licensee personnel). In addition, the licensee shall include the relocated information in the Updated Final Safety Analysis Report submitted to the NRC, pursuant to 10 CFR 50.71(e), as was described in the licensee's application dated October 2, 1996, and evaluated in the staff's safety evaluation dated July 13, 1998.

FOR THE NUCLEAR REGULATORY COMMISSION

William Reckley

William Reckley, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: July 13, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 191

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Revise the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. *The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
V	V
VI	VI
XI	XI
*XII	XII
3/4 3-28	3/4 3-28
3/4 3-29	-
3/4 3-30	-
3/4 3-31	-
3/4 3-32	-
3/4 3-33	-
3/4 3-34	-
3/4 3-35	-
3/4 3-42	3/4 3-42
*3/4 3-57	3/4 3-57
3/4 3-58	-
*3/4 7-17	3/4 7-17
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6-16	6-16

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

INDEX

BASES

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TABLE 4.3-13

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Gross Radioactivity Monitor(s) (provides alarm and automatic isolation)				
a. Liquid Radwaste Effluents Line	D*	P**	R	Q
2. Flow Monitor(s)				
a. Liquid Radwaste Effluent Line	D*	N/A	R	N/A

* During releases via this pathway

** A SOURCE CHECK is not required if the background activity is greater than the activity of the check source.

3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING AND AIR FILTRATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6.1 Two independent control room emergency air conditioning and air filtration systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one control room emergency air conditioning or air filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 Each control room emergency air conditioning system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Starting each unit from the control room, and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature $\leq$ 84°F D.B.

b. At least once per 18 months by verifying a system flow rate of 9900 cfm $\pm$ 10%.

4.7.6.1.2 Each control room emergency air filtration system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

3. Verifying a system flow rate of 2000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 2000 cfm ±10%.

2. Verifying that on a control room high radiation test signal, the system automatically isolates the control room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ±10%.

3/4.7.11 FIRE BARRIERS

LIMITING CONDITION FOR OPERATION

DELETED

3/4.7.12 SPENT FUEL POOL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.7.12 The structural integrity of the spent fuel pool shall be maintained in accordance with Specification 4.7.12.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

a. With the structural integrity of the spent fuel pool not conforming to the above requirements, in lieu of any other report, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days of a determination of such non-conformity.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.12.1 Inspection Frequencies - The structural integrity of the spent fuel pool shall be determined per the acceptance criteria of Specification 4.7.12.2 at the following frequencies:

a. At least once per 92 days after the pool is filled with water. If no abnormal degradation or other indications of structural distress are detected during five consecutive inspections, the inspection interval may be extended to at least once per 5 years.

b. Within 24 hours following any seismic event which actuates or should have actuated the seismic monitoring instrumentation.

4.7.12.2 Acceptance Criteria - The structural integrity of the spent fuel pool shall be determined by a visual inspection of at least the interior and exterior surfaces of the pool, the struts in the tilt pit, the surfaces of the separation walls, and the structural slabs adjoining the pool walls. This visual inspection shall verify no changes in the concrete crack patterns, no abnormal degradation or other signs of structural distress (i.e, cracks, bulges, out of plumbness, leakage, discolorations, efflorescence, etc.).

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

The PURGE as defined in the definitions section is a release under a purge permit, whereas continuous ventilation is defined as operation of the purge system after the requirements of the purge permit have been satisfied. When securing the containment purge system to meet the ACTION requirements of this Specification, at least one supply valve and one exhaust valve is to be closed, and the supply and exhaust fans secured.

3/4.3.3.2 DELETED

3/4.3.3.3 DELETED

3/4.3.3.4 DELETED

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

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

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations."

The Reactor Vessel Level Monitor is provided as a means of indicating level in the reactor vessel during accident conditions. A minimum of two operable level sensors in the upper plenum region and one operable level sensor in the dome region are required for RVLMS channel operability. When Reactor Coolant Pumps are running, all except the dome sensors are interlocked to read "invalid" due to flow induced variables that may offset the sensor outputs. If the equipment is inaccessible due to health and industrial safety concerns (for example, high radiation area, low oxygen content of the containment atmosphere) or due to physical location of the fault (for example, probe failure in the reactor vessel), then operation may continue until the next scheduled refueling outage and a report filed.

measurement assurance activities with NBS. These standards permit calibrating the system over its intended range of energy and measurement range. For subsequent CHANNEL CALIBRATION, sources that have been related to the initial calibration are used.

ANNUAL REPORTS[1]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5. Reports required on an annual basis shall include:

a. A tabulation on an annual basis for the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[2] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received form external sources shall be assigned to specific major work functions.

b. The complete results of steam generator tube inservice inspections performed during the report period (reference Specification 4.4.5.5.b).

c. Documentation of all challenges to the pressurizer safety valves.

d. A diesel generator data report which provides the number of valid tests and the number of valid failures for each diesel generator.

e. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded the results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history

1/ 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.

2/ This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office no later than the 15th of each month following the calendar month covered by the report.

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. Deleted

c. Deleted

d. Deleted

e. Inoperable Fire Detection Instrumentation

f. Inoperable Fire Suppression Systems

g. Deleted

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 192 AND 191 TO

FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NOS. 1 AND 2

DOCKET NOS. 50-313 AND 50-368

1.0 INTRODUCTION

By letter dated October 2, 1996, Entergy Operations, Inc. (the licensee) submitted a request for changes to the Arkansas Nuclear One, Units 1 and 2 (ANO-1&2), Technical Specifications (TSs). The requested changes would revise the ANO-1&2 TSs by relocating selected TS requirements related to instrumentation from the TS to the Updated Final Safety Analysis Report (UFSAR). The Nuclear Regulatory Commission (NRC) provided guidance to all holders of operating licenses or construction permits for nuclear power reactors on the proposed TS changes in Generic Letter 95-10, "Relocation of Selected Technical Specifications Requirements Related to Instrumentation," dated December 15, 1995.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls.

Section 50.36 requires limiting conditions for operation which meet any of the following criteria to be in the TS:

(1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

(2) A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(3) A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; and

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(4) A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

As a result, existing TS requirements which fall within or satisfy any of the criteria must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents.

3.0 EVALUATION

The requested amendments would relocate selected TS requirements related to instrumentation from the TS to the Technical Requirements Manual (TRM). The TRM is a part of the UFSAR. The relocated requirements for ANO-1 include the limiting conditions for operation and related surveillance requirements for:

TS 3.5.1.10	Control Room Ventilation Chlorine Detection System
TS 3.5.1.13	Seismic Monitoring Instrumentation
Table 3.5.1-1	Instrumentation Limiting Conditions for Operation Item 9 - Chlorine Detection System Item 14 - Seismic Monitoring Instrumentation Note 27 - Seismic Monitoring Instrumentation Action Statement
Table 4.1-1	Instrumentation Surveillance Requirements Item 31 - Turbine Overspeed Trip Mechanism Item 42 - Seismic Monitoring Instruments Item 52 - Control Room Chlorine Detector

The relocated requirements for ANO-2 include the following limiting conditions for operation and related surveillance requirements:

TS 3/4.3 3.3	Seismic Instrumentation (Including Tables 3.3.7 and 4.3.4)
TS 3/4.3.3.4	Meteorological Instrumentation (Including Tables 3.3.8 and 4.3.5)
TS 3/4.3.3.7	Chlorine Detection Systems
TS 3/4..3.4	Turbine Overspeed Protection
TS 3/4.7.6(d)	Delete Surveillance for Control Room Isolation on High Chlorine Test Signal

The staff's evaluation of the various proposed relocations are provided below:

SEISMIC INSTRUMENTATION
(ANO-1 TS 3.5.1.13, Table 3.5.1-1, and Table 4.1-1; ANO-2 TS 3/4.3.3.3)

Section VI(a)(3) of Appendix A to 10 CFR Part 100, requires that seismic monitoring instrumentation be provided to promptly determine the response of those nuclear power plant features important to safety in the event of an earthquake. This capability is required to allow for a comparison of the measured response to that used in the design basis for the unit. Comparison of such data is needed to (1) determine whether the plant can continue to be operated safely, and (2) permit such timely action as may be appropriate. The requirements do not address the need for seismic monitoring instrumentation that would automatically shut down the plant when an earthquake occurs which exceeds a predetermined intensity. The licensee has proposed to relocate these provisions to the UFSAR such that future changes to the operation and surveillance of the seismic monitoring instrumentation could be changed under 10 CFR 50.59.

The capability of the plant to withstand a seismic event or other design-basis accident is determined by the initial design and construction of systems, structures, and components. The instrumentation is used to alert operators to the seismic event and evaluate the plant response. The Final Policy Statement explained that instrumentation to detect precursors to reactor coolant pressure boundary leakage, such as seismic instrumentation, is not included in the first criterion. As discussed above, the seismic instrumentation does not serve as a protective design feature or part of a primary success path for events which challenge fission product barriers.

Accordingly, the staff has concluded that the requirements for seismic monitoring instrumentation do not meet the criteria in 10 CFR 50.36. The limiting conditions for operation and surveillance requirements for seismic monitoring instrumentation were removed from the latest versions of the standard technical specifications for all reactor types (including Standard Technical Specifications Babcock and Wilcox Plants, NUREG-1430, and Standard Technical Specifications for Combustion Engineering Plants, NUREG-1432).

CHLORINE DETECTION SYSTEM
(ANO-1 TS 3.5.1.10, Table 3.5.1-1, and Table 4.1-1; ANO-2 TS 3/4.3.3.7)

Chlorine detection systems ensure that sufficient capability is available to promptly detect and initiate protective action to isolate the control room in the event of an accidental chlorine release. Normal plant operation could be hampered in the event of the accidental release of chlorine from onsite or offsite sources. Staff positions regarding the relationship of the chlorine detection systems to the general design criteria (GDC) appear in NUREG-0800, "Standard Review Plan" (SRP); Regulatory Guide (RG) 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"; and RG 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release." The licensee has proposed to relocate these provisions to the UFSAR such that future changes to the operation and surveillance of the chlorine detection system could be made under 10 CFR 50.59.

As discussed above, chlorine detection systems may serve an important role in the protection of control room personnel from internal or external hazards related to toxic gases. However, the release of chlorine or other hazardous chemicals is not part of an initial condition of a design basis accident or transient analysis that assumes a failure of or presents a challenge to the integrity of a fission product barrier. Since the release of toxic gases is not assumed to initiate or occur simultaneously with design basis accidents or transients involving challenges to fission product barriers, the chlorine detection system is not part of a success path for the mitigation of those accidents or transients.

Accordingly, the staff has concluded that the requirements for chlorine [toxic gas] detection systems do not meet the criteria in 10 CFR 50.36. The limiting conditions for operation and surveillance requirements for chlorine detection systems were removed from the latest versions of the standard technical specifications for all reactor types (including Standard Technical Specifications Babcock and Wilcox Plants, NUREG-1430, and Standard Technical Specifications for Combustion Engineering Plants, NUREG-1432).

TURBINE OVERSPEED PROTECTION
(ANO-1 TS Table 4.1-1, ANO-2 TS 3/4.3.4)

General Design Criterion 4 of Appendix A to 10 CFR Part 50 requires that structures, systems, and components important to safety be appropriately protected from the effects of missiles that may result from equipment failures. The turbine is equipped with control valves and stop valves which control turbine speed during normal plant operation and protect it from overspeed during abnormal conditions. The turbine overspeed protection system consists of separate mechanical and electrical sensing mechanisms which are capable of initiating fast closure of the steam valves. Currently, ANO-1 TS require a test of the turbine overspeed trip mechanism on a refueling outage frequency and ANO-2 TS 3/4.3.4 requires particular operability and surveillance requirements for these steam control and stop valves to minimize the potential for fragment missiles that might be generated as the result of a turbine overspeed event. The licensee has proposed to relocate these provisions to the TRM such that future changes to the operation and surveillance of the turbine overspeed features could be changed under 10 CFR 50.59.

Although the design basis accidents and transients include a variety of system failures and conditions which might result from turbine overspeed events and potential missiles striking various plant systems and equipment, the system failures and plant conditions are much more likely to be caused by events other than turbine failures. In view of the low likelihood of turbine missiles, assumptions related to the turbine overspeed protection system are not part of an initial condition of a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. The turbine overspeed protection system is not relied upon in the design basis accident or transient analyses as a primary success path which functions or actuates to mitigate such events. Probabilistic safety assessments and operating experience have demonstrated that proper maintenance of the turbine overspeed control valves is important to minimize the potential for overspeed events and turbine damage; however that experience has also demonstrated that there is low likelihood of significant risk to public health and safety because of turbine overspeed events. Further, the potential for and consequences of turbine overspeed events are diminished by the favorable orientation of the turbine, relative to the likely path of any turbine missiles, and the licensee's inservice inspection program, which must comply with 10 CFR 50.55(a), and a surveillance program for the turbine control and stop valves derived from the manufacturer's recommendations.

Accordingly, the staff has concluded that the requirements for turbine overspeed protection do not meet the 10 CFR 50.36 criteria for inclusion in the TS and may be relocated to the UFSAR, where future changes to the requirements may be made under 10 CFR 50.59. The limiting conditions for operation and surveillance requirements for turbine overspeed protection were not included in the latest versions of the standard technical specifications for all reactor types (including Standard Technical Specifications Babcock and Wilcox Plants, NUREG-1430, and Standard Technical Specifications for Combustion Engineering Plants, NUREG-1432).

METEOROLOGICAL MONITORING INSTRUMENTATION (ANO-2 TS 3/4.3.3.4)

The meteorological monitoring instrumentation is used to measure environmental parameters (wind direction, speed, and air temperature differences) which may affect the distribution of radioactive effluents following a release of radioactive material. In 10 CFR 50.47, "Emergency Plans," and 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," the Commission requires power plant licensees to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Timely access to accurate local meteorological data is important for estimating potential radiation doses to the public and for determining appropriate protective measures. In 10 CFR 50.36a(a)(2), the Commission requires nuclear power plant licensees to submit annual reports specifying the quantity of each of the principal radionuclides released to unrestricted areas in liquid and airborne effluents and such other information as may be required by the NRC to estimate maximum potential annual radiation doses to the public. A knowledge of meteorological conditions in the vicinity of the reactor is important in providing a basis for estimating annual radiation doses resulting from radioactive materials released in airborne effluents. Accordingly, the meteorological monitoring instrumentation serves a useful function in estimating radiation doses to the public from either routine or accidental releases of radioactive materials to the atmosphere. The licensee has proposed to relocate these provisions to the UFSAR such that future changes to the operation and surveillance of the meteorological monitoring instrumentation could be changed under 10 CFR 50.59.

Although the meteorological monitoring instrumentation serves a useful function as described above, it does not serve a primary protective function so as to warrant inclusion in the TS in accordance with the criteria of the final policy statement. The instrumentation does not serve to ensure that the plant is operated within the bounds of initial conditions assumed in design basis accident and transient analyses or that the plant will be operated to preclude transients or accidents. Likewise, the meteorological instrumentation does not serve as part of the primary success path of a safety sequence analysis used to demonstrate that the consequences of these events are within the appropriate acceptance criteria.

Accordingly, the staff has concluded that the requirements for meteorological monitoring instrumentation do not meet the 10 CFR 50.36 criteria, so they may be removed from the TS and relocated to the UFSAR. The limiting conditions for operation and surveillance requirements for meteorological monitoring instrumentation were not included in the latest versions of the standard technical specifications for all reactor types (including Standard Technical Specifications for Combustion Engineering Plants, NUREG-1432).

In summary, these specific instrumentation requirements are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to public health and safety. Further, they do not fall within any of the four criteria set forth in the Commission's Final Policy Statement and subsequently incorporated into 10 CFR 50.36. In addition, the Staff finds that sufficient regulatory controls exist under 10 CFR 50.59 to address future changes to these requirements. Accordingly, the staff has concluded that these requirements may be relocated from the TS to the licensee's UFSAR.

The licensee has included several changes to other TS that are affected by the relocation of the above requirements. The staff has reviewed these changes and, in light of accepting the relocation to the updated FSAR of the limiting conditions for operation and surveillance requirements for the subject instrumentation, finds the proposed changes to be appropriate. In addition, the licensee has proposed to relocate those TS Bases affected by the relocation of the TS requirements. The staff finds the relocation of the affected TS Bases to be acceptable.

The NRC has included as a condition for the approval of this request that the relocated requirements will be included in the appropriate submittal of the updated FSAR for ANO-1 and ANO-2 in accordance with 10 CFR 50.71(e).

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (62 FR 2188). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: W. D. Reckley

Date: July 13, 1998