FEBRUARY 1981

Docket No. 50-313

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Mr. William Cavanaugh, III
Vice President, Generation
and Construction
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

The Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications and adds two license conditions in response to your application dated May 16, 1979, as supplemented by letters dated June 8, 1979 and February 12, 1980; your application dated June 6, 1979, also supplemented February 12, 1980; and your application dated October 31, 1980. During our review of your proposed amendments, we found certain modifications were necessary to meet our requirements. Your staff has agreed to these modifications and they have been incorporated in this amendment. Outstanding items of your February 12, 1980 letter will be the subject of future correspondence.

This amendment modifies the ANO-1 Appendix A Technical Specifications dealing with the emergency feedwater system and other TMI-2 Lessons Learned Category "A" issues. This amendment also adds license conditions relating to a Systems Integrity Measurements Program and an improved Iodine Measurement capability.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original signed by
Robert W. Reid
Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

8102170420

RECEIVED
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Enclosures:
1. Amendment No. 50 to DPR-51
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

concur in amendment and Fed. Reg. Notice.

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DATE	1/2?/81	1/27/81	1/24/81	1/26/81	1/29/81		

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

February 2, 1981

Docket No. 50-313

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Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: ARKANSAS NUCLEAR ONE UNIT 1

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Amendment No. 50
  Referenced documents have been provided PDR

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

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NRC FORM 102 (1-76)

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

Docket
50-313

February 2, 1981

Docket No. 50-313

Mr. William Cavanaugh, III
Vice President, Generation
and Construction
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

The Commission has issued the enclosed Amendment No. 50 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). This amendment consists of changes to the Technical Specifications and adds two license conditions in response to your application dated May 16, 1979, as supplemented by letters dated June 8, 1979 and February 12, 1980; your application dated June 6, 1979, also supplemented February 12, 1980; and your application dated October 31, 1980. During our review of your proposed amendments, we found certain modifications were necessary to meet our requirements. Your staff has agreed to these modifications and they have been incorporated in this amendment. Outstanding items of your February 12, 1980 letter will be the subject of future correspondence.

This amendment modifies the ANO-1 Appendix A Technical Specifications dealing with the emergency feedwater system and other TMI-2 Lessons Learned Category "A" issues. This amendment also adds license conditions relating to a Systems Integrity Measurements Program and an improved Iodine Measurement capability.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 50 to DPR-51
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

Arkansas Power & Light Company

cc w/enclosure(s):

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Manager, Licensing
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Mr. Nicholas S. Reynolds
Debevoise & Liberman
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Washington, DC 20036

Arkansas Polytechnic College
Russellville, Arkansas 72801

Honorable Ermil Grant
Acting County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

Mr. Paul F. Levy, Director
Arkansas Department of Energy
3000 Kavanaugh
Little Rock, Arkansas 72205

Director, Criteria and Standards
Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region VI Office
ATTN: EIS COORDINATOR
1201 Elm Street
First International Building
Dallas, Texas 75270

cc w/enclosure(s) & incoming dtd.:
5/16/79, 6/6 & 6/8/79, 2/12/80, 10/31/80

Director, Bureau of Environmental
Health Services
4815 West Markham Street
Little Rock, Arkansas 72201

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE - UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No.

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

   A. The applications for amendment by Arkansas Power and Light Company (the licensee) dated May 16, 1979, as supplemented June 8, 1979, and February 12, 1980; June 6, 1979, as supplemented February 12, 1980; and October 31, 1980, comply 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 applications, 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.

8102170424

2. Accordingly, Facility Operating License No. DPR-51 is hereby amended by revising paragraph 2.c.(2) and adding paragraphs 2.c.(6) and 2.c.(7) as follows and by changing the Technical Specifications as indicated in the attachment to this license amendment:

2.c.(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 50 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

2.c.(6) Systems Integrity

The licensee shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

2.c.(7) Iodine Monitoring

The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: February 2, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 50

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
5	5
21	21
22	22
40	40
--	40a (new page)
41	41
43a	43a
45b	45b
--	45c (new page)
69	69
70*	70*
72	72
72a	72a
--	72b (new page)
73	73
105	105
--	105a (new page)
117	117
118	118

* Overleaf page; no change.

1.7 REACTOR BUILDING

Reactor building integrity exists when the following conditions are satisfied:

a. The equipment hatch is closed and sealed and both doors of the personnel lock and emergency lock are closed and sealed, or b. below.

b. At least one door on each of the personnel lock and emergency lock is closed and sealed during personnel access or repair.

c. All non-automatic reactor building isolation valves and blind flanges are closed as required.

d. All automatic reactor building isolation valves are operable or deactivated in the closed position.

e. The reactor building leakage determined at the last testing interval satisfies Specification 4.4.1.

1.8 FIRE SUPPRESSION WATER SYSTEM

The fire suppression water system consists of: water sources, pumps, and distribution piping with associated sectionalizing isolation valves. Such valves include the hose standpipe shutoff valves and the first valve ahead of the water flow alarm device or each sprinkler system.

1.9 STAGGERED TEST BASIS

A staggered test basis shall consist of:

a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or designated component at the beginning of each subinterval.

3.1.3 Minimum Conditions For Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply.

3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent $\Delta k/k$ until a steam bubble is formed and an indicated water level between 45 and 305 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality.

3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours.

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent $\Delta k/k$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated $\Delta k/k$ would limit the magnitude of power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below the limits of Figure 3.1.2-2 provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than one (1) percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.

The requirement that 2 of the 3 emergency-powered pressurizer heaters be operable provides assurance that sufficient heater capacity ($\geq$ 126 kw) is available to provide reactor coolant system pressure control during a loss of off-site power.

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.5

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the turbine cycle components for removal of reactor decay heat.

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.

Specifications

3.4.1 The reactor shall not be heated, above 280F unless the following conditions are met:

1. Capability to remove a decay load of 5% full reactor power by at least one of the following means:

   a. A condensate pump and a main feedwater (MFW) pump, using turbine by-pass valve.

   b. A condensate pump and the auxiliary feedwater (AFW) pump using turbine by-pass valve.

2. Fourteen of the steam system safety valves are operable.

3. A minimum of 16.3 ft. (107,000 gallons) of water is available in the condensate storage tank.

4. Both emergency feedwater (EFW) pumps and both EFW block valves are capable of automatic actuation, or a dedicated operator is available for their operation.*

5. Both main steam block valves and both main feedwater isolation valves are operable.

6. The emergency feedwater valves associated with Specification 3.4.1.4 shall be operable.

3.4.2 The Steam Line Break Instrumentation and Control System (SLBIC) shall be operable when main steam pressure exceeds 700 psig and shall be set to actuate at 600 ± 25 psig.

* One train of EFW may be removed from the control-grade automatic actuation mode for purposes of surveillance testing of the automatic actuation circuitry for a period not to exceed one (1) hour per test without invoking the reporting requirements of Specification 6.12.3.

3.4.3 Components required by Specification 3.4.1 and 3.4.2 to be operable shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 and 3.4.2 within 24 hours, the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specification 3.4.1 and 3.4.2 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours.

3.4.4 The reactor shall not be heated above 280F unless both EFW pumps are operable.

3.4.5 If the condition specified in 3.4.4 cannot be met:

1. With one EFW flow path inoperable, the unit shall be brought to hot shutdown within 36 hours, and if not restored to an operable status within the next 36 hours, the unit shall be brought to cold shutdown within the next 12 hours or at the maximum safe rate.

2. If both EFW trains are inoperable, the AFW pump shall be demonstrated operable immediately, and the unit shall be brought to hot shutdown within one hour. The unit shall be placed in cold shutdown within the next 12 hours or at the maximum safe rate.

3. If both EFW trains and the AFW pump are inoperable, the unit shall be immediately run back to $\leq 5\%$ full power with feedwater supplied from the MFW pumps. As soon as an EFW train or the AFW train is operable, the unit shall be placed in cold shutdown within the next 12 hours or at the maximum safe rate.

Bases

The feedwater flow required to remove decay heat corresponding to 5% full power with saturated steam at 1065 psia (lowest setting of steam safety valve) as a function of feedwater temperature is:

Feedwater Temperature	Flow
60	758
90	777
120	799
140	814

The feedwater system and the turbine bypass system are normally used for decay heat removal and cooldown above 280F. Feedwater makeup is supplied by operation of a condensate pump and either a main or the auxiliary feedwater pump.

In the incredible event of loss of all AC power, feedwater is supplied by the turbine driven emergency feedwater pump which takes suction from the condensate storage tank. Decay heat is removed from a steam generator by steam relief through the atmospheric dump valves or safety valves. Fourteen of the steam system safety valves will relieve the necessary amount of steam for rated reactor power.

The minimum amount of water in the condensate storage tank would be adequate for about 4.5 hours of operation. This is based on the estimate of the average emergency flow to a steam generator being 390 gpm. This operation time with the volume of water specified would not be reached, since the decay heat removal system would be brought into operation within 4 hours or less.

If the turbine driven emergency feedwater pump has not been verified to be operable within 3 months prior to heatup its operability will be verified upon reaching hot shutdown conditions.

The SLBIC System is designed to isolate the steam generators to assure that only one steam generator will experience uncontrolled blowdown following a steam line break. Normal steam line operating pressures are approximately 900 psig at all power levels, thus operability above 700 psig with actuation at 600 $\pm$25 psig are appropriate. The setpoint is based on severe transients in the main steam lines resulting in rapid pressure decays.

The control-grade EFW automatic actuation system is required per NUREG-0578 to assure that EFW will be available when necessary. This control-grade system is fully testable, but only at the risk of cold EFW reaching a hot steam generator during operation. To reduce the risk of this, and the subsequent transient, the EFW train to be tested may be removed from the automatic actuation mode if the other train is operable by automatic action, and the train to be tested is still operable is the manual mode.

References

FSAR, Section 10

for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of both the RPS and the ESAS enable complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

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 Steam Line Break Instrumentation and Control System (SLBIC) is designed to automatically close the Main Steam Block valves and the Main Feedwater Isolation valves upon loss of pressure in either of the two main steam lines.

The SLBIC is also designed to be reset from its trip position only when the system is shut down or the Main Steam line pressure is below 650 psig.

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 recommendations 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."

REFERENCE

FSAR, Section 7.1

Table 3.5.1-1 (Cont.)

OTHER SAFETY RELATED SYSTEMS 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. Steam line break instrumentation control system (SLBIC).					
(a) SLBIC Control & Logic Channels	2	1	2	1	Notes 9, 5
3. Pressurizer level channels	3	N/A	2	1	Note 10
4. Emergency Feedwater flow channels	2/S.G.	N/A	1	0	Note 10
5. RCS subcooling margin monitors	2	N/A	1	0	Note 10
6. Electromatic relief valve flow monitor	2	N/A	1	0	Note 11
7. Electromatic relief block valve position indicator	1	N/A	1	0	Note 12
8. Pressurizer code safety valve flow monitors	2/valve	N/A	1/valve	0	Note 10

Notes:

1. Initiate a shutdown using normal operating instructions and place the reactor in the hot shutdown condition if the requirements of Columns 3 and 4 are not met within 12 hours.

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

3. When 1 of 2 intermediate range instrument channels is greater than 10^{-10} amps, hot shutdown is not required.

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies.

5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours.

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, Specification 3.3 shall apply.

7. These channels initiate control rod withdrawal inhibits not reactor trips at <10% rated power. Above 10% rated power, these inhibits are bypassed.

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies.

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.

10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies.

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.

Table 4.1-1
Instrument Surveillance Requirements

	Channel Description	Check	Test	Calibrate	Remarks
1.	Protective Channel Coincidence Logic	NA	M	NA	
2.	Control Rod Drive Trip Breaker	NA	M(1)	NA	(1) To include tripping of breakers via shunt trip circuit.
3.	Power Range Amplifier	NA	NA	T/W(1)	(1) Heat balance calibration twice weekly under steady state operating conditions, daily under non-steady state operating conditions.
4.	Power Range Channel	S M(1)	M	M(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers monthly and after each startup if not done previous week.
5.	Intermediate Range Channel	S	P/M	NA	
6.	Source Range Channel	S(1)	P	NA	(1) When in service.
7.	Reactor Coolant Temperature Channel	S	M	R	
8.	High Reactor Coolant Pressure Channel	S	M	R	
9.	Low Reactor Coolant Pressure Channel	S	M	R	
10.	Flux-Reactor Coolant Flow Comparator	S	M	R	
11.	Reactor Coolant Pressure Temperature Comparator	S	M	R	
12.	Pump Flux Comparator	S	M	R	

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
13. High Reactor Building Pressure Channel	S	M	R	
14. High Pressure Injection Logic Channel	NA	M	NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S	M (1)	R	(1) Including test of shutdown bypass function (ECCS bypass function).
b. Reactor Building 4 psig Channel	S	M	R	
16. Low Pressure Injection Logic Channel	NA	M	NA	
17. Low Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S	M (1)	R	(1) Including test of shutdown bypass function (ECCS bypass function).
b. Reactor Building 4 psig Channel	S	M	R	
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	M	NA	
19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
a. Reactor Building 4 psig Channels	S	M	R	

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
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) Shall Also Be Tested During Refueling Shutdown Prior to Repressurization at a pressure greater than 300 but less than 420 psig.
31. Turbine Overspeed Trip Mechanism	NA	R	NA	
32. Steam Line Break Instrumentation and Control System Logic Test & Control Circuits	W	Q	R	
33. Diesel Generator Protective Relaying Starting Interlocks And Circuitry	M	Q	NA	
34. Off-site Power Undervoltage And Protective Relaying Interlocks And Circuitry	W	R	R	
35. Borated Water Storage Tank Level Indicator	W	NA	R	
36. Reactor Trip Upon Loss of Main Feedwater Circuitry	NA	PC	NA	

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
37. Boric Acid Addition Tank				
a. Level Channel	NA	NA	R	
b. Temperature Channel	M	NA	R	
38. Deleted				
39. Sodium Hydroxide Tank Level Indicator	NA	NA	R	
40. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning
41. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check
42. Reactor Trip Upon Turbine Trip Circuitry	NA	PC	NA	
43. Strong Motion Acceleographs	Q(1)	NA	Q	(1) Battery Check
44. ESAS Manual Trip Functions				
a. Switches & Logic	NA	P	NA	
b. Logic	NA	M	NA	
45. Reactor Manual Trip	NA	P	NA	
46. Reactor Building Sump Level	NA	NA	R	

Table 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks
47.	EFW Actuation Control Logic	NA	M	R	
48.	EFW Flow Indication	R	NA	R	
49.	RCS subcooling margin monitor	D	NA	R	
50.	Electromatic relief valve flow monitor	D	NA	R	
51.	Electromatic relief block valve position indicator	D	NA	R	
52.	Pressurizer safety valve flow monitor	D	NA	R	
53.	Pressurizer water level indicator	D	NA	R	

Note:

S-Each Shift	T/W-Twice per Week	R-Once every 18 months
D-Daily	B/M-Every 2 Months	NA-Not applicable
W-Weekly	Q-Quarterly	PC-Prior to Going Critical if Not Done Within Previous 31 Days
M-Monthly	P-Prior to Each Startup if Not Done Previous Week	

Table 4.1-2

Minimum Equipment Test Frequency

	Item	Test	Frequency
1.	Control Rods	Rod Drop Times of all Full Length Rods 1/	Each Refueling Shutdown
2.	Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions
3.	Pressurizer Code Safety Valves	Setpoint	One Valve Every 18 Months
4.	Main Steam Safety Valves	Setpoint	Four Valves Every 18 Months
5.	Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown
6.	Reactor Coolant System Leakage	Evaluate	Daily
7.	Emergency-powered Pressurizer Heaters	Power availability	Daily
		Heater capacity functional test	Every 18 Months
8.	Reactor Building Isolation Trip	Functioning	Every 18 Months
9.	Service Water Systems	Functioning	Every 18 Months
10.	Spent Fuel Cooling System	Functioning	Every 18 Months when irradiated fuel is in the pool.
11.	Decay Heat Removal System Isolation Valve Automatic Closure and Isolation System	Functioning	Every 18 Months

1/ Same as tests listed in Section 4.7

4.8 EMERGENCY FEEDWATER PUMP

<u>Applicability</u>

Applies to the periodic testing of the turbine and electric motor driven emergency feedwater pumps.

<u>Objective</u>

To verify that the emergency feedwater pump and associated valves are operable.

<u>Specification</u>

4.8.1 Each EFW train shall be demonstrated operable:

a) By verifying on a STAGGERED TEST BASIS:

1) at least once per 31 days or upon achieving hot shutdown following a plant heatup and prior to criticality, that the turbine-driven pump starts, operates for a minimum of 5 minutes, and develops a discharge pressure of $\geq$ 1560 psig through the automatically isolable recirculation flow path.

2) at least once per 31 days by verifying that the motor driven EFW pump starts, operates for a minimum of 5 minutes and develops a discharge pressure of $\geq$ 1400 psig on minimum recirculation flowpath.

b) at least once per 31 days on a STAGGERED TEST BASIS by verifying that each valve (manual, power operated or automatic) in each EFW flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

c) prior to exceeding 280F Reactor Coolant temperature and after any EFW system alignment alterations by verifying that each manual valve in each EFW flowpath which, if mis-positioned may degrade EFW operation, is locked in its correct position.

d) at least once per 92 days on a STAGGERED TEST BASIS by cycling each motor-operated valve in each flowpath through at least one complete cycle.

e) at least once per 18 months by functionally testing each EFW train and:

1) Veryifying that each automatic valve in each flowpath actuates automatically to its correct position on receipt of an actuation signal.

2) Verifying that the automatic steam supply valves associated with the steam turbine driven EFW pump actuate to their correct positions upon receipt of an actuation signal.

3) Verifying that the motor-driven EFW pump starts automatically upon receipt of an actuation signal.

4) Verifying that feedwater is delivered to each steam generator using the electric motor-driven EFW pump.

5) Verifying that the EFW system can be operated manually by over-riding ICS actuation signals to the EFW valves.

Bases

The monthly testing frequency will be sufficient to verify that both emergency feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps. The cycling of the emergency valves assures valve operability when called upon to function.

The functional test, performed once every 18 months, will verify that the flow path to the steam generators is open and that water reaches the steam generators from the emergency feedwater system. The test is done during shutdown to avoid thermal cycle to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater.

The automatic actuation circuitry testing and calibration will be performed per Surveillance Specification 4.1, and will be sufficient to assure that this circuitry will perform its intended function when called upon.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2.-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2.-2. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable position, except for (1) the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for fire protection training shall be maintained and shall meet or exceed the requirements of Section 27 of the NFPA Code-1975 with the exception of frequency of training which shall be six times per year.

6.5 REVIEW AND AUDIT

6.5.1 Plant Safety Committee (PSC) Function

6.5.1.1 The Plant Safety Committee shall function to advise the General Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Safety Committee shall be composed of the:

(See Page 121)

Table 6.2-1

ARKANSAS NUCLEAR ONE

MINIMUM SHIFT CREW COMPOSITION #

UNIT 1

LICENSE CATEGORY	ABOVE COLD SHUTDOWN	COLD AND REFUELING SHUTDOWNS
SOL	1	1*
OL	2	1
NON-LICENSED	2	1
SHIFT TECHNICAL ADVISOR	1	None required

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising refueling operations after the initial fuel loading.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

Additional Requirements:

1. At least one licensed Operator shall be in the control room when fuel is in the reactor.

2. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

3. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

4. All refueling operations after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

5. At least 5 individuals with fire protection training shall be maintained onsite at all times. These individuals shall not include the minimum shift crew necessary for safe shutdown of the unit (2 members) or any personnel required for other essential functions during a fire emergency.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 50 TO

FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

Introduction

By letter dated May 16, 1979, as supplemented by letters dated June 8, 1979 and February 12, 1980; by letter dated June 6, 1979, also supplemented February 12, 1980; and by letter dated October 31, 1980, Arkansas Power and Light Company (the licensee) proposed changes to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1. The letter dated February 12, 1980, which revised in total the letters dated May 16 and June 6 and 8, 1979, proposed changes to the TSs relating to the emergency feedwater system. This was in reference to IE Bulletins 79-05A and 79-05B and the request of our letter dated May 31, 1979, which discussed the licensee's implementation of the Commission's May 17, 1979 Order. These proposed changes to the TSs would also incorporate certain TMI-2 Lessons Learned Category "A" requirements which were requested by our letter dated July 2, 1980. The licensee's letter dated October 31, 1980, was in direct response to our letter dated July 2, 1980.

Background

By our letter dated September 13, 1979, we issued to all operating nuclear power plants requirements established as a result of our review of the TMI-2 accident. Certain of these requirements, designated Lessons Learned Category "A" requirements, were to have been completed by the licensee prior to any operation subsequent to January 1, 1980. Our evaluation of the licensee's compliance with these Category "A" items was attached to our letter to the licensee dated March 10, 1980.

In order to provide reasonable assurance that operating reactor facilities are maintained within the limits determined acceptable following the implementation of the TMI-2 Lessons Learned Category "A" items, we requested that licensees amend their TSs to incorporate additional Limiting Conditions of Operation and Surveillance Requirements, as appropriate. This request was transmitted to all licensees on July 2, 1980. Included therein were model specifications that we had determined to be acceptable.

The licensee, by letter dated February 12, 1980, submitted proposed changes to the TSs relating to the emergency feedwater system that would be responsive to the auxiliary feedwater concerns of our letter dated July 2, 1980. By letter dated February 4, 1980, the licensee submitted proposed TS changes related to diverse containment isolation that would also be responsive to the containment isolation concerns of our letter dated July 2, 1980. This subject was a part of another licensing action. The licensee's application dated October 31, 1980, relates to other TMI-2 Lessons Learned Category "A" requirements and is a direct response to our request of July 2, 1980. Each of the issues identified by the NRC staff and the licensee's submittal is discussed in the evaluation below.

Evaluation

Emergency Power Supply Requirements

The pressurizer water level indicators, pressurizer relief and block valves, and pressurizer heaters are important in a post-accident situation. Adequate emergency power supplies add assurance of post-accident functioning of these components. The licensee has the requisite emergency power supplies. The licensee has proposed adequate TSs which provide for a daily channel check and an 18-month channel calibration and actions in the event of component inoperability. We have reviewed these proposed TSs and find that the emergency power supplies are reasonably ensured for post-accident functioning of the subject components and are thus acceptable.

Direct Indication of Flow Valve Position

The licensee has provided direction indication of flow downstream of the PORV and safety valves in the control room. These indications are a diagnostic aid for the plant operator and provide no automatic action. The licensee has provided TSs with a daily channel check and an 18-month channel calibration requirement; thus, the TSs are acceptable and they meet our July 2, 1980 model TS criteria.

Instrumentation for Inadequate Core Cooling

The licensee has installed an instrument system to detect the effects of low reactor coolant level and inadequate core cooling. These instruments, subcooling meters, receive and process data from existing plant instrumentation. We previously reviewed this system in our Safety Evaluation dated March 10, 1980. The licensee submitted TSs with a daily channel check and an 18-month channel calibration requirement and actions to be taken in the event of component inoperability. We conclude the TSs are acceptable as they meet our July 2, 1980 model TS criteria.

Auto Initiation and Operability of Emergency Feedwater System

The licensee has provided for the automatic initiation of emergency feedwater flow on loss of normal feedwater flow. We have previously reviewed the design and installation of this system as part of the requirements of the Commission's Confirmatory Order dated May 17, 1979 (our letter dated January 18, 1980). The circuits are designed to be testable and the design retains the capability of manual actuation from the control room even in the event of failure of the auto-initiating circuitry. The TSs submitted by the licensee by letter dated February 12, 1980, provide for operability requirements, limiting conditions of operation and appropriate action in the event of component inoperability, and surveillance requirements for system and components. We conclude that the proposed TSs are acceptable.

Emergency Feedwater Flow Indications

The licensee has installed emergency feedwater flow indicators that conform to safety grade requirements. We reviewed this system in our Safety Evaluation dated March 10, 1980. The licensee has proposed operability requirements by letter dated October 31, 1980, and surveillance requirements by letter dated February 12, 1980. We have reviewed the proposed TSs and find them acceptable.

Shift Technical Advisor (STA)

Our request indicated that the TSs related to minimum shift manning should be revised to reflect the augmentation of an STA. The licensee's application would add one STA to each shift to perform the function of accident assessment. The individual performing this function will have at least a bachelor's degree or equivalent in a scientific or engineering discipline with special training in plant design, and response and analysis of the plant for transients and accidents. Part of the STA duties are related to operating experience review function. Based on our review, we find the licensee's submittal to satisfy our requirements and is acceptable.

Integrity of Systems Outside Containment

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to a Systems Integrity Measurements Program. Such a condition would require the licensee to effect an appropriate program to eliminate or prevent the release of significant amounts of radioactivity to the environment via leakage from engineered safety systems and auxiliary systems, which are located outside reactor containment. We discussed this with the licensee's staff and the licensee's staff agreed to adopt such a license condition; accordingly we have included this condition in the license.

Iodine Monitoring

Our letter dated July 2, 1980, indicated that the license should be amended by adding a license condition related to iodine monitoring. Such a condition would require the licensee to effect a program which would ensure the capability to determine the airborne iodine concentration in areas requiring personnel access under accident conditions. We discussed this with the licensee's staff and the licensee's staff agreed to adopt such a license condition; accordingly, we have included this condition in the license.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because that amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: February 2, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCEKT NO. 50-313

ARKANSAS POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 50 to Facility Operating License No. DPR-51, issued to Arkansas Power and Light Company, which revised the license and the Technical Specifications for operation of the Arkansas Nuclear One, Unit No. 1 (the facility) located in Pope County, Arkansas. The amendment is effective as of its date of issuance.

The amendment modifies the Technical Specifications dealing with the emergency feedwater system and other TMI-2 Lessons Learned Category "A" issues. This amendment also adds license conditions relating to a Systems Integrity Measurements Program and an improved Iodine Measurement capability.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated May 16, 1979 (as supplemented June 8, 1979 and February 12, 1980), June 6, 1979 (as supplemented February 12, 1980), and October 31, 1980, (2) Amendment No. 50 to License No. DPR-51, and (3) the Commission's related Safety Evaluation. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. and at the Arkansas Polytechnic College, Russellville, Arkansas. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 2nd day of February 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Licensing