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Docket No. 50-313

Mr. William Cavanaugh, III
Senior Vice President,
Energy Supply
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Cavanaugh:

The Commission has issued the enclosed Amendment No. 61 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated June 6, 1979, as supplemented February 12, 1980, January 29, 1981, and May 29, 1981.

This amendment modifies the ANO-1 Appendix A Technical Specifications by adding operability requirements, limiting conditions for operation, and surveillance and test requirements for the Anticipatory Reactor Trip System on loss of main feedwater and/or turbine trip.

With the issuance of this amendment, we consider Item II.K.2.10, "Safety Grade Anticipatory Reactor Trip" of the TMI Action Plan resolved for ANO-1.

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

CP
1

Sincerely,

Original signed by

Guy S. Vissing, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 61 to DPR-51
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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no legal object.

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DATE	10/21/81	10/22/81	10/22/81	10/28/81	10/28/81	10/27/81	

Arkansas Power & Light Company

cc w/enclosure(s):

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Little Rock, Arkansas 72203

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

Regional Radiation Representative
EPA Region VI
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cc w/enclosure(s) & incoming dtd.:

6/6/79, 2/12/80, 1/29/81 & 5/29/81

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. 61
License No. DPR-51

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated June 6, 1979, as supplemented February 12, 1980, January 29, 1981, and May 29, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 61, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: November 3, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 61

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.

42a

43a

43b (new page)

44a (new page)

45d

72

72a

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

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

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

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

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

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

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

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

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.

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip trip may be bypassed up to 20% to allow sufficient margin for bringing the turbine on line at approximately 15%.

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 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 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'd)

REACTOR PROTECTION SYSTEM (Cont'd)

	1	2	3	4	5
Functional Unit	No. of channels	No. of channels for system trip	Min. operable channels	Min. degree of redundancy	Operator action if conditions of column 3 or 4 cannot be met
11. Reactor trip upon loss of Main Feedwater	4	2	2	1	Notes 1, 15
12. Reactor trip upon turbine trip	4	2	2	1	Notes 1, 16

Table 3.5.1-1 (Cont'd)

Notes Cont'd.

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

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

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

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

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 (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 loads to Startup Transformer No. 2.
35. Borated Water Storage Tank Level Indicator	W	NA	R	
36. Reactor Trip Upon Loss of Main Feedwater Circuitry	M	PC	R	

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.	Degraded Voltage Monitoring	W	R	R	
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	M	PC	R	
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	

Amendment No. ~~25~~, ~~39~~, ~~50~~, ~~60~~, 61

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 61 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

Introduction

By letter dated June 6, 1979 (Reference 1), supplemented by letters dated February 12, 1980, January 29, 1981, and May 29, 1981, (References 2, 3 and 4, respectively), Arkansas Power and Light Company (the licensee or AP&L) requested amendment to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The proposed amendment would modify the Technical Specifications (TSs) by adding operability requirements, limiting conditions for operation, and surveillance and test requirements relating to the Anticipatory Reactor Trip System (ARTS) on loss of main feedwater and/or turbine trip.

Background

In response to our Order of May 17, 1979 (Reference 5), the licensee implemented a control grade ARTS and submitted a preliminary design of the safety grade ARTS by letter dated October 8, 1979 (Reference 6). We approved the preliminary design of the ARTS by letter dated December 20, 1979 (Reference 7), and the licensee subsequently submitted the final design of the ARTS by letter dated August 8, 1980 (Reference 8). Our Safety Evaluation (SE) on the final design was issued December 16, 1980 (Reference 9).

Evaluation

The proposed TSs include a number of channels for system trip, minimum operable channels and total number of sensors. Minimum degree of redundancy is also listed, and the limiting conditions for operation are stated for cases where channel operability and redundancy requirements are not met. Reactor power level for bypass of ARTS is listed as 10% Full Power (FP) and 20% FP for loss of main feedwater pump and turbine trip respectively. Functional testing of the ARTS would be performed on a monthly basis whereas the channel calibration and verification would be carried out during refueling outages.

The ARTS sensors, their number, independence and redundancy, frequency of functional test and channel calibration in the proposed TS change, comply with the NRC staff approved figures in our SE (Reference 9).

The power level figure in the proposed TS (for bypass of ARTS for loss of main feedwater pump) is the correct value (10% FP), whereas 20% FP value in the SE (Reference 9) was based on an error in the licensee's submittal (Reference 10).

Based on our review of the proposed TSs in line with our approved modification and SE (Reference 9), we have determined that the proposed modifications to the ANO-1 TSs for ARTS would not involve a decrease in the margin of safety or an increase in the probability of consequences of accidents and are, therefore, acceptable.

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 this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the 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.

Dated: November 3, 1981

References

1. Letter, W. Cavanaugh (AP&L) to Director of Nuclear Reactor Regulation, (NRC), June 6, 1979, Proposed TSs.

2. Letter, W. Cavanaugh (AP&L) to Director of Nuclear Reactor Regulation, (NRC), February 12, 1980, Proposed TSs.

3. Letter, W. Cavanaugh (AP&L) to Director of Nuclear Reactor Regulation, (NRC), January 29, 1981, Proposed TSs for the ARTS.

4. Letter, W. Cavanaugh (AP&L) to Director of Nuclear Reactor Regulation, (NRC), May 29, 1981, TS Change Request for Degraded Grid Voltage and ARTS.

5. NRC Order for providing ARTS for loss of main feedwater and/or turbine trip, ANO-1, Docket No. 50-313, May 17, 1979.

6. Letter, D. Trimble (AP&L) to R. W. Reid (NRC), October 8, 1979, Conceptual Design of Safety Grade ART.

7. Letter, R. Reid (NRC) to W. Cavanaugh (AP&L), December 20, 1979, SE for Preliminary Design of Safety Grade ARTS.

8. Letter, D. C. Trimble (AP&L) to Director of Nuclear Reactor Regulation, (NRC), August 8, 1980, ARTS.

9. Letter, R. Reid (NRC) to W. Cavanaugh (AP&L), December 16, 1980, SE for Final Design of Safety Grade ARTS.

10. Letter, D. C. Trimble (AP&L) to Director of Nuclear Reactor Regulation, NRC, February 5, 1981, Additional Information on ARTS.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET 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. 61 to Facility Operating License No. DPR-51, issued to Arkansas Power and Light Company (the licensee), which revised the Technical Specifications for operation of Arkansas Nuclear One, Unit No. 1 (ANO-1) located in Pope County, Arkansas. The amendment is effective as of the date of issuance.

The amendment modifies the ANO-1 Appendix A Technical Specifications by adding operability requirements, limiting conditions for operation, and surveillance and test requirements for the Anticipatory Reactor Trip System on loss of main feedwater and/or turbine trip.

The application for the amendment complies 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 licensee's application dated June 6, 1979, as supplemented February 12, 1980, January 29, 1981, and May 29, 1981, (2) Amendment No. 61 to License No. DPR-51, and (3) the Commission's related Safety Evaluation. All of 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 Tech University, 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 3rd day of November 1981.

FOR THIS NUCLEAR REGULATORY COMMISSION

John F. Stolz

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