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

JUNE 16 1981

Docket No. 50-313

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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. 57 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1. This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 28, 1980. During our review of your proposed amendment, we found that certain modifications were necessary to meet our requirements. Your staff has agreed to these modifications and they have been incorporated in this amendment.

The amendment revises the TSs by providing a redefinition of the term OPERABLE and the addition of general Limiting Conditions for Operation.

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

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLTZ"
John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 57 to DPR-51
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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Concur in license amendment and Fed. Reg. Notice

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SURNAME	RIngram	GVissing;cf	JStolz	TNovak	M. Rothschild		
DATE	6/3/81	6/3/81	6/4/81	6/9/81	6/09/81		

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Docket file

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June 16, 1981

Docket No. 50-313

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Sincerely,

Gary S. Vissing for

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

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See next page

Arkansas Power & Light Company

cc w/enclosure(s):

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

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.:
11/28/80

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. 57
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 November 28, 1980, 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.

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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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 57, 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: June 16, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 57

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.

Pages	
2	52*
15a (new page)	53d
15b (new page)	54
15c (new page)	55
16	59
18	59a
18a (new page)	59b
19	66
21	66b
27	66g
28*	66m
30	66n
32	66o
46	66p
49	66q
51	66r

*No change on these pages, included for document completeness only.

refueling temperature (normally 140F). Pressure is defined by Specification 3.1.2. A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.

1.2.7 Refueling Operation

An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.

1.2.8 Startup

The reactor shall be considered in the startup mode when the shutdown margin is reduced with the intent of going critical.

1.3 OPERABLE - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

1.4 PROTECTION INSTRUMENTATION LOGIC

1.4.1 Instrument Channel

An instrument channel is the combination of sensor, wires, amplifiers and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control and/or protection. An instrument channel may be either analog or digital.

1.4.2 Reactor Protection System

The reactor protection system is shown in Figures 7-1 and 7-9 of the FSAR. It is that combination of protective channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protective trip breakers and activating relays or coils.

A protection channel, as shown in Figure 7-1 of the FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply

3. LIMITING CC_ITIONS FOR OPERATION

3.0 LIMITING CONDITION FOR OPERATION (GENERAL)

3.0.1 The Limiting Conditions for Operation requirements shall be applicable during the Reactor Operating Conditions or other conditions specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, no further actions need be taken.

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated Action requirements, within one hour action shall be initiated to place the unit in an OPERATING CONDITION in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the Action requirements, the Action may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

3.0.4 Entry into an Operating Condition or other specified applicability condition shall not be made unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the Action statements unless otherwise excepted. This provision shall not prevent passage through Operating Conditions as required by a Limiting Condition for Operation.

3.0.5 When a system, subsystem, train, component or device is determined to be inoperable solely because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE; or likewise satisfy the requirements of this specification. Unless both conditions (1) and (2) are satisfied, within 2 hours action shall be initiated to place the unit in an OPERATING CONDITION in which the applicable Limiting Condition for Operation does not apply by placing it, as applicable, in:

1. At least HOT STANDBY within the next 6 hours,
2. At least HOT SHUTDOWN within the following 6 hours, and
3. At least COLD SHUTDOWN within the subsequent 24 hours.

This Specification is not applicable in Cold Shutdown or Refueling Shutdown.

3.0.1 This specification defines the applicability of each specification in terms of defined Operating Conditions or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation.

3.0.3 This specification delineates the Action to be taken for circumstances not directly provided for in the Action statements and whose occurrence would violate the intent of the specification.

3.0.4 This specification provides that entry into an Operating Condition or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the Limiting Condition for Operation statements of the appropriate specifications.

3.0.5 This specification delineates what additional conditions must be satisfied to permit operation to continue when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the Limiting Condition for Operation statements associated with individual systems, subsystems, trains, components or devices to be consistent with the Limiting Condition for Operation statements of the associated electrical power source. It allows operation to be governed by the time limits of the Limiting Condition for Operation for the normal or emergency power source, not the individual Limiting Condition for Operation statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.7.2.C provides for a 2 day out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable Action statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the Limiting Condition for Operation statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components and devices must be OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e., be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, shutdown is required in accordance with this specification.

As a further example, Specification 3.7.1.A requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be OPERABLE. Specification 3.7.2.B provides a 24 hour out-of-service time when both required offsite circuits are not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite circuits would also be inoperable. This would dictate invoking the applicable Limiting Condition for Operation statements for each of the applicable LCOs. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the Limiting Condition for Operation statement for the inoperable normal power sources instead, provided the other specified conditions are satisfied. In this case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be OPERABLE, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, shutdown is required in accordance with this specification.

During Cold Shutdown and Refueling Shutdown, Specification 3.0.5 is not applicable and thus the individual Action statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

3.1 REACTOR CO[illegible]NT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

3.1.1 Operational Components

Specification

3.1.1.1 Reactor Coolant Pumps

A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1.

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. With no reactor coolant pumps or decay heat removal pumps running, immediately suspend all operations involving a reduction of boron concentration in the reactor coolant system.

3.1.1.2 Steam Generator

A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280F.

3.1.1.3 Pressurizer Safety Valves

A. Both pressurizer code safety valves shall be operable when the reactor is critical. With one pressurizer code safety valve operable, either restore the valve to operable status within 15 minutes or be in Hot Shutdown within 12 hours.

B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable.

3.1.1.4 Reactor Internals Vent Valves

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable.

3.1.1.5 Reactor Coolant Loops

A. With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable:

3.1.2 Pressurization, Heatup and Cooldown Limitations

Specification

3.1.2.1 Hydro Tests

For thermal steady state system hydro tests, the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core, under the provisions of 3.1.2.3, and to ASME Code limits when no fuel assemblies are present provided the reactor coolant system limits are to the right of and below the limit line in Figure 3.1.2-1. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.1.2.2 Leak Tests

Leak tests required by Specification 4.3 shall be conducted under the provision of 3.1.2.3. The provisions of Specification 3.0.3 are not applicable.

3.1.2.3 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-2 and 3.1.2-3, and are as follows:

Heatup:

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-2. The heatup rates shall not exceed those shown in Figure 3.1.2-2.

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the right of and below the limit line in Figure 3.1.2-3. Cooldown rates shall not exceed those shown in Figure 3.1.2-3.

3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is above 100F.

3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430F.

3.1.2.6 With the limits of Specifications 3.1.2.3 or 3.1.2.4 or 3.1.2.5 exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS Tavg and pressure to less than 200F and 500 psia, respectively, within the following 30 hours.

3.1.2.7 Prior to reaching six effective full power years of operation, Figures 3.1.2-1, 3.1.2-2 and 3.1.2-3 shall be updated for the next service period in accordance with 10CFR50, Appendix G, Section V.B. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with Specification 4.2.7. The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.8. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.1.2.8 The updated proposed technical specifications referred to in 3.1.2.7 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR Part 50, Appendix G, Section V.C.

Bases

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.(1) These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rate of 100F per hour satisfies stress limits for cyclic operation.(2) The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100F satisfies stress levels for temperatures below the DTT.(3)

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in BAW-1440(4).

Figures 3.1.2-1, 3.1.2-2, and 3.1.2-3 present the pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown respectively. The limit curves are applicable through the fifth effective full power year of operation. These curves are adjusted by 25 psi and 10F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-2 for reactor criticality and on Figure 3.1.2-1 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

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.

3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in 15 minutes or be in at least Hot Shutdown within the next 15 minutes.

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 a power excursion resulting from a reduction of moderator density.

3.1.6 Leakage

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (exceeding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

3.1.6.3 If it is determined that any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system strength boundary (such as the reactor vessel, piping, valve body, etc., except steam generator tubes), the reactor shall be shutdown and a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

3.1.6.4 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, or 3.1.6.3, the rate of cooldown and the conditions of shutdown shall be determined by the safety evaluation for each case and reported as required by Specification 6.12.3.

3.1.6.5 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude of the leak, shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10CFR20.

3.1.6.6 If reactor shutdown is required by Specification 3.1.6.1, 3.1.6.2, or 3.1.6.3, the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.7 When the reactor is at power operation, three reactor coolant leak detection systems of different operating principles shall be in operation. One of these systems is sensitive to radioactivity and consists of a radioactive gas detector and an air particulate activity detector. Both of these instruments may be out-of-service simultaneously for a period of no more than 72 hours provided two other means are available to detect leakage and reactor building air samples are taken and analyzed in the laboratory at least once per shift; otherwise, be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.1.6.8 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1 and 3.1.6.6 except that such losses when added to leakage shall not exceed 30 gpm.

3.1.6.9 If the reactor coolant system pressure isolation valve leakage is greater than the values given in Table 3.1.6.9, isolate (by having at least two valves in the high pressure piping closed*) the high pressure portion of the affected system from the low pressure portion within 4 hours and apply Specification 3.3.6, or be in at least hot shutdown within the next 6 hours and in cold shutdown within the following 30 hours.

Bases

Every reasonable effort will be made to reduce reactor coolant leakage, including evaporative losses (which may be on the order of 0.5 gpm), to prevent a large leak from masking the presence of a smaller leak. Reactor building sump level, water inventory balances, radiation monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactive contamination and cleanup or it could develop into a still more serious problem; and therefore, the first indication of such leakage will be followed up as soon as practicable.

Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.

When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Operating Staff and will be documented in writing and approved by the Superintendent. Under these conditions, an allowable reactor coolant system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also available even during a loss of off-site power.

If leakage is to the reactor building it may be identified by one or more of the following methods:

a. Leakage is monitored by a level indicator in the reactor building sump. Changes in normal sump level may be indicative of leakage from any of the systems located inside the reactor building such as the reactor coolant system, service water system, intermediate cooling system and steam and feedwater lines or condensation of humidity within the reactor building atmosphere. The reactor building sump contains 63.6 gallons per inch of height. A 1 gpm leak would be detected in less than 1 hour.

*The motor operated valve shall remain closed and power supplies deenergized.

3.1.7 Moderator Temperature Coefficient of Reactivity

Specification

3.1.7.1 The moderator temperature coefficient (MTC) shall be non-positive whenever thermal power is $\geq$95% of rated thermal power and shall be less positive than 0.5×10^{-4} $\Delta k/k/°F$ whenever thermal power is <95% of rated thermal power and the reactor is not shutdown.

3.1.7.2 The MTC shall be determined to be within its limits by confirmatory measurements prior to initial operation above 5% of rated thermal power after each fuel loading. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the limits in 3.1.7.1 above.

3.1.7.3 With the MTC outside any one of the above limits, be in at least Hot Standby within 6 hours.

Bases

A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power, the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.5 \times 10^{-4}$ $\Delta k/k/°F$ corrected to 95% of rated power. All other accident analyses as reported in the FSAR have been performed for a range of moderator temperature coefficients including $+0.5 \times 10^{-4}$ $\Delta k/k/°F$.

3.1.9 Control Rod Operation

Specification

3.1.9.1 The concentration of dissolved gases in the reactor coolant shall be limited to 100 std. cc/liter of water at the reactor vessel outlet temperature.

3.1.9.2 Allowable combinations of pressure and temperature for control rod operation shall be to the left of and above the limiting pressure versus temperature curve for a dissolved gas concentration of 100 std. cc/liter of water as shown in Figure 3.1.9-1.

3.1.9.3 In the event the limits of Specifications 3.1.9.1 or 3.1.9.2 are exceeded, the center control rod drive mechanism shall be checked for accumulation of undissolved gases. The temperature, pressure and dissolved gas concentration shall be restored to within their limits within 24 hours or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

Bases

By maintaining the reactor coolant temperature and pressure as specified above, any dissolved gases in the reactor coolant system are maintained in solution.

Although the dissolved gas concentration is expected to be approximately 20-40 std. cc/liter of water, the dissolved gas concentration is conservatively assumed to be 100 std. cc/liter of water at the reactor vessel outlet temperature.

The limiting pressure versus temperature curve for dissolved gases is determined by the equilibrium pressure versus temperature curve for the dissolved gas concentration of 100 std. cc/liter of water. The equilibrium total pressure is the sum of the partial pressure of the dissolved gases plus the partial pressure of water at a given temperature. The margin of error consists of the maximum pressure difference between the pressure sensing tap and lowest pressure point in the system, the maximum pressure gage error, and the pressure difference due to the maximum temperature gage error.

If either the maximum dissolved gas concentration (100 std. cc/liter of water) is exceeded or the operating pressure falls below the limiting pressure versus temperature curve, the center CRDM should be checked for accumulation of undissolved gases.

3.5.2 Control Rod Group and Power Distribution Limits

Applicability

This specification applies to power distribution and operation of control rods during power operation.

Objective

To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.

Specification

3.5.2.1 The available shutdown margin shall be not less than 1% $\Delta k/k$ with the highest worth control rod fully withdrawn. With the shutdown margin less than 1%$\Delta k/k$, immediately initiate and continue boration injection until the required shutdown margin is restored.

3.5.2.2 Operation with inoperable rods:

1. Operation with more than one inoperable rod, as defined in Specification 4.7.1 and 4.7.2.3, in the safety or regulating rod groups shall not be permitted.

2. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of 1%$\Delta k/k$ available shutdown margin. Boration may be initiated either to the worth of the inoperable rod or until the regulating and transient rod groups are withdrawn to the limits of Specification 3.5.2.5.3, whichever occurs first. Simultaneously, a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.

3. If within one (1) hour of the determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that a 1%$\Delta k/k$ available shutdown margin exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the Hot Standby condition until this margin is established.

4. Following the determination of an inoperable rod as defined in Specification 4.7.1, all remaining rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.

5. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the thermal power allowable for the reactor coolant pump combination.

3.5.3 Safety Features Actuation System Setpoints

Applicability

This specification applies to the safety features actuation system actuation setpoints.

Objective

To provide for automatic initiation of the safety features actuation system in the event of a breach of reactor coolant system integrity.

Specification

The safety features actuation setpoints and permissible bypasses shall be as follows:

Functional Unit	Action	Setpoint
High Reactor Building Pressure*	Reactor Building Spray	$\leq$30 psig (44.7 psia)
	High Pressure Injection	$\leq$4 psig (18.7 psia)
	Start of Reactor Building Cooling and Reactor Building Isolation	$\leq$4 psig (18.7 psia)
	Reactor Bldg. Ventilation	$\leq$4 psig (18.7 psia)
	Low Pressure Injection	$\leq$4 psig (18.7 psia)
	Penetration Room Ventilation	$\leq$4 psig (18.7 psia)
Low Reactor Coolant System Pressure**	High Pressure Injection	$\geq$ 1500 psig
	Low Pressure Injection	$\geq$ 1500 psig
	Start of Reactor Building Cooling and Reactor Building Isolation	$\geq$ 1500 psig

*May be bypassed during reactor building leak rate test.
**May be bypassed below 1750 psig and is automatically reinstated above 1750 psig.

With the safety features actuation setpoints less conservative than the above values, declare the channel inoperable and apply the applicable Action requirements of Table 3.5.1-1 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the trip setpoint value.

3.5.4 Incore Instrumentation

Applicability

Applies to the operability of the incore instrumentation system.

Objective

To specify the functional and operational requirements of the incore instrumentation system.

Specification

Above 80 percent of operating power determined by the reactor coolant pump combination (Table 2.3-1) at least 23 individual incore detectors shall be operable to check gross core power distribution and to assist in the periodic calibration of the out-of-core detectors in regard to the core imbalance trip limits. The detectors shall be arranged as follows and may be a part of both basic arrangements.

3.5.4.1 Axial Imbalance

A. Three detectors, one in each of three strings, shall lie in the same axial plane with one plane in each axial core half.

B. The axial planes in each core half shall be symmetrical about the core mid-plane.

C. The detector shall not have radial symmetry.

3.5.4.2 Radial Tilt

A. Two sets of four detectors shall lie in each core half. Each set of four shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

B. Detectors in the same plane shall have quarter core radial symmetry.

With the incore detector system inoperable, do not use the system for the above applicable monitoring function. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Bases

A system of 52 incore flux detector assemblies with 7 detectors per assembly has been provided primarily for fuel management purposes. The system includes data display and record functions and is also used for out-of-core nuclear instrumentation calibration and for core power distribution verification.

A. The out-of-core nuclear instrumentation calibration includes:

1. Calibration of the split detectors at initial reactor startup, during the power escalation program, and periodically thereafter.

2. A comparison check with the incore instrumentation in the event one of the four out-of-core power range detector assemblies gives abnormal readings during operation.

3. Confirmation that the out-of-core axial power splits are as expected.

B. Core power distribution verification includes:

1. Measurement at low power initial reactor startup to check that power distribution is consistent with calculations.

2. Subsequent checks during operation to insure that power distribution is consistent with calculations.

3. Indication of power distribution in the event that abnormal situations occur during reactor operation.

C. The safety of unit operation at or below 80 percent of operating power[1] for the reactor coolant pump combinations without the core imbalance trip system has been determined by extensive 3-D calculations. This will be verified during the physics startup testing program.

D. The minimum requirement for 23 individual incore detectors is based on the following:

1. An adequate axial imbalance indication can be obtained with 9 individual detectors. Figure 3.5.4-1 shows a typical set of three detector strings with 3 detectors per string that will indicate an axial imbalance. The three detector strings are the center one, one from the inner ring of symmetrical strings and one from the outer ring of symmetrical strings.

2. Figure 3.5.4-2 shows a typical detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from 2 detectors in a radial quadrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

3. Figure 3.5.4-3 combines Figures 3.5.4-1 and 3.5.4-2 to illustrate a typical set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions. Startup testing will verify the adequacy of this set of detectors for the above functions.

E. At least 23 specified incore detectors will be operable to check power distribution above 80 percent power determined by reactor coolant pump combination. These incore detectors will be read out either on the computer or on a recorder. If a set of 23

3.5.5 Fire Detection Instrumentation

Applicability

This specification applies to fire detection instrumentation utilized within fire areas containing safety related equipment or circuitry for the purposes of protecting that safety related equipment or circuitry.

Objective

To provide immediate notification of fires in areas where there exists a potential for a fire to disable safety related systems.

Specification

3.5.5.1 A minimum of 50% of the heat/smoke detectors in the locations specified in Table 3.5-5 shall be operable.

3.5.5.2 If less than 50% of the fire detectors in any of the locations designated in Table 3.5-5 are operable, within one hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour and restore the equipment to operable status within 14 days or prepare and submit a report to the Commission pursuant to Specification 6.12.3.2(b) within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the instrument(s) to operable status. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Bases

The various detectors provide alarms that notify the operators of the existence of a fire in its early stages thus providing early initiation of fire protection. The detectors in the main and auxiliary control rooms also provide automatic fire protection initiation.

The detectors required to be operable in the various areas represent one half of those installed.

Operability of the fire detection instrumentation ensures that operable warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected area(s) is required to provide detection capability until the inoperable instrumentation is restored to operability.

3.6 REACTOR BUILDING

Applicability

Applies to the integrity of the reactor building.

Objective

To assure reactor building integrity.

Specification

3.6.1 Reactor building integrity shall be maintained whenever all three (3) of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200F or greater.

c. Nuclear fuel is in the core.

Without reactor building integrity, restore the integrity within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.6.2 Reactor building integrity shall be maintained when the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. The provisions of Specification 3.0.3 are not applicable.

3.6.3 Positive reactivity insertions which would result in the reactor being subcritical by less than 1% $\Delta k/k$ shall not be made by control rod motion or boron dilution whenever reactor building integrity is not in force. The provisions of Specification 3.0.3 are not applicable.

3.6.4 The reactor shall not be taken critical or remain critical if the reactor building internal pressure exceeds 3.0 psig or a vacuum of 5.5 inches Hg. With the reactor critical, restore the containment pressure to within its limits within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.6.5 Prior to criticality following a refueling shutdown, a check shall be made to confirm that all manual reactor building isolation valves which should be closed are closed and locked, as required. The provisions of Specification 3.0.3 are not applicable.

3.6.6 If, while the reactor is critical, a reactor building isolation valve is determined to be inoperable in a position other than the closed position, the other reactor building isolation valve (except for check valves) in the line shall be tested to insure operability. If the inoperable valve is not restored within 48 hours, the reactor shall be brought to the cold shutdown condition within an additional 24 hours or the operable valve will be closed.

Bases

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence there will be no pressure buildup in the reactor building if the reactor coolant system ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110°F and the building is subsequently cooled to an internal temperature of less than 50°F.

When reactor building integrity is established, the limits of 10 CFR 100 will not be exceeded should the maximum hypothetical accident occur.

References

FSAR, Section 5

3.8.6 During the handling of irradiated fuel in the reactor building, at least one door on the personnel and emergency hatches shall be closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed.

3.8.8 When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.10 The reactor building purge isolation system, including the radiation monitors, shall be tested and verified to be operable within 7 days prior to refueling operations. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.12 All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.13 No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill Type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.14 For the maximum fuel pool heat load capacity (i.e., seven reload batches - 413 assemblies - stored in the pool at the time of discharge of the full core), the full core to be discharged shall be cooled in the reactor vessel a minimum of 175 hours prior to discharge. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.15 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.16 The spent fuel shipping cask shall not be carried by the Auxiliary Building crane pending the evaluation of the spent fuel cask drop accident and the crane design by AP&L and NRC review and approval. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

BASES:

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition.

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.[1]

The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.

The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.[2] The boron concentration will be maintained above 1,800 ppm. Although this concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.

Specification 3.8.11 is required as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.[3]

Specification 3.8.14, which requires cooling of the full core for 175 hours prior to discharge to the spent fuel pool when seven reload batches are already stored in the pool, is necessary to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded.

Specification 3.8.15 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates, until the review specified in 3.8.16 is completed.

Specification 3.8.16 assures that the spent fuel cask drop accident cannot occur prior to completion of the NRC staff's review of this potential accident and the completion of any modifications that may be necessary to preclude the accident or mitigate the consequences. Upon satisfactory completion of the NRC's review, Specification 3.8.16 shall be deleted.

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

3.10 SECONDARY SYSTEM ACTIVITY

Applicability

Applies to the limiting conditions of secondary system activity for operation of the reactor.

Objective

To limit the maximum secondary system activity.

Specification

The I-131 dose equivalent of the radioiodine activity in the secondary coolant shall not exceed 0.17 uCi/gm. With the secondary coolant activity in excess of 0.17 uCi/gm I-131, be in at least Hot Standby within 6 hours and in Cold Shutdown within the following 30 hours.

Bases

For the purpose of determining a maximum allowable secondary coolant activity, the activity contained in the mass released following the rupture of a steam generator tube, a steam line break outside containment and a loss of load incident were considered.

The whole body dose is negligible since any noble gases entering the secondary coolant system are continuously vented to the atmosphere by the condenser vacuum pumps. Thus, in the event of a loss of load incident or steam line break, there are only small quantities of these gases which would be released.

The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for Specification 3.1.4.1 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 uCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for Specification 3.1.4.1. For the less probable accident of a steam line break, the assumption is made that a loss of 1×10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 uCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident.

3.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability

Applies to byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

3.12.1 The source leakage test performed pursuant to Specification 4.14 shall be capable of detecting the presence of 0.005 uCi of radioactive material on the test sample. If the test reveals the presence of 0.005 uCi or more of removable contamination, it shall immediately be withdrawn from use, decontaminated and repaired, or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 uCi or less of beta and/or gamma emitting material or 5 uCi or less of alpha emitting material. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.12.2 A special report shall be prepared and submitted to the Commission within 90 days if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

3.12.3 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

3.15 FUEL HANDLING AREA VENTILATION SYSTEM

Applicability

Applies to the operability of the fuel handling area ventilation system.

Objective

To ensure that the fuel handling area ventilation system will perform within acceptable levels of efficiency and reliability.

Specification

3.15.1 The fuel handling area ventilation system shall be in operation whenever irradiated fuel handling operations are in progress in the fuel handling area of the auxiliary building and shall have the following performance capabilities:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows ($\pm$ 10%) on HEPA filters and charcoal adsorber banks shall show $\geq$ 99% DOP removal and $\geq$99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$ 90% radioactive methyl iodide removal at a velocity within $\mp$ 20% of system design, 0.05 to 0.15 mg/m^3 inlet methyl iodide concentration, $\geq$ 70% R. H. and $\geq$ 125F.

c. Fans shall be shown to operate within $\pm$ 10% design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate ($\pm$ 10%).

e. Air distribution shall be uniform within $\pm$ 20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system.

3.15.2 If the requirements of Specification 3.15.1 cannot be met, irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed). The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Bases

The fuel handling area ventilation system is designed to filter the auxiliary building atmosphere during fuel handling operations to limit the release of activity should a fuel handling accident occur. The system consists of one circuit containing two exhaust fans and a filter train. The fans are redundant and only one is required to be operating. The filter train consists of a prefilter, a HEPA filter and a charcoal adsorber in series.

3.17 FIRE SUPPRESSION WATER SYSTEM

Applicability

This specification applies to the portions of the fire suppression water system necessary to provide fire protection to safety related equipment.

Objective

To assure that fire suppression is available to safety related equipment.

Specification

3.17.1 The fire suppression water system shall be operable at all times with two high pressure pumps, each with a capacity of at least 2500 gpm, with their discharges aligned to the fire suppression header, and with an operable flow path capable of transferring water through distribution piping with operable sectionalizing control valves to the shutoff valve ahead of each hose standpipe and the water flow alarm device on each sprinkler system.

3.17.2 With one pump inoperable, restore the inoperable equipment to operable status within seven days or prepare and submit a report to the Commission pursuant to Specification 6.12.3.2(b) within the next 30 days outlining the plans and procedures to be used to provide for the loss of redundancy in this system.

3.17.3 With the fire suppression water system inoperable:

a. Establish a backup fire suppression water system within 24 hours; and

b. submit a report in accordance with Specification 6.12.3.1(b).

c. If "a." above cannot be fulfilled, place the reactor in Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

Bases

The fire pumps supply the only source of water for all fire suppression systems utilizing water. Each pump is individually capable of providing full flow required for proper fire suppression water system operation. The pumps start automatically on low system pressure.

In the event that the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a 24-hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued operation of the plant.

3.18 FIRE SUPPRESSION SPRINKLER SYSTEM

Applicability

This specification applies to the following fire suppression sprinkler systems protecting safety related areas:

a. Each of the four reactor building cable penetration areas.

b. Each of the four cable penetration rooms.

c. Each of the two emergency diesel generator rooms.

d. Cable spreading room.

e. Each of the two diesel generator fuel vaults.

f. Hallway El 372 (Zone 98-J).

g. Condensate demineralizer area.*

Objective

To assure that fire suppression is available to safety related equipment located in the above-listed areas.

Specification

3.18.1 The above-listed sprinkler systems shall be operable at all times.

3.18.2 With one or more of the above-listed sprinkler systems inoperable, establish a continuous fire watch (or operable smoke and/or heat detection equipment with control room alarm) with backup fire suppression equipment for the applicable area(s) within one hour. Restore the system(s) to operable status within 14 days or prepare and submit a report to the Commission pursuant to Specification 6.12.3.2(b) within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system(s) to operable status. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Bases

Safety related equipment located in various areas is protected by sprinkler systems. The operability of these systems ensures that adequate fire suppression capability is available to confine and extinguish a fire occurring in the applicable areas. In the event a system is inoperable, alternate backup fire fighting equipment or operable detection equipment is required to be made available until the inoperable equipment is restored to service.

*To be implemented no later than July 30, 1979.

3.19 CONTROL ROOM AND AUXILIARY CONTROL ROOM HALON SYSTEMS

Applicability

This specification applies to the Halon systems utilized as the fire suppression system for the control room and auxiliary control room.

Objective

To assure that fire suppression is available to the safety related equipment in the control room and auxiliary control room.

Specification

3.19.1 The three control room and auxiliary control room Halon systems shall be operable at all times with the storage tanks having at least 95% of full charge weight and 90% of full charge pressure.

3.19.2 With any of the control room and auxiliary control room Halon systems inoperable, establish backup fire suppression equipment for the affected area within one hour. Restore the system(s) to operable status within 14 days or prepare and submit a report to the Commission pursuant to Specification 6.12.3.2(b) within 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Bases

Safety related circuitry located in portions of the control room and auxiliary control room is protected by the Halon systems. The operability of these systems ensures that adequate fire suppression capability is available to confine and extinguish a fire occurring in the control room or the auxiliary control room.

In the event that the system(s) is inoperable, alternate backup fire fighting equipment is required to be made available in the affected area until the inoperable equipment is restored to service.

3.20 FIRE HOSE STATIONS

Applicability

This specification applies to the fire hose stations protecting areas containing safety related equipment.

Objective

To assure that manual fire suppression capability is available to all safety related equipment.

Specification

3.20.1 All fire hose stations protecting areas containing safety related equipment shall be operable whenever the equipment in these areas is required to be operable.

3.20.2 With one or more of the fire hose stations of 3.20.1 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an operable hose station within one hour. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Bases

The operability of the fire hose stations adds additional assurance that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located.

In the event that portions of this system are inoperable, backup fire hose equipment is required to be made available in the affected area(s) until the inoperable equipment is restored to service.

3.21 PENETRATION FIRE BARRIERS

Applicability

This specification applies to penetration fire barriers relied on for restriction of fire damage such that safety related equipment in areas other than the main fire area are not affected.

Objective

To assure that penetration fire barriers protecting safety related areas perform their separation function.

Specification

3.21.1 All penetration fire barriers protecting safety related areas shall be intact at all times.

3.21.2 With one or more of the required penetration fire barriers not intact, establish a continuous fire watch (or operable smoke and/or heat detection equipment with control room alarm) on at least one side of the affected penetration within one hour. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

Bases

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility fire protection program and are subject to periodic inspections.

During periods of time when the barriers are not functional, a continuous fire watch or operable detection equipment is required to be maintained in the vicinity of the affected barrier until the barrier is restored to functional status.

3.22 REACTOR BUILDING PURGE FILTRATION SYSTEM

Applicability

This specification applies to the operability of the reactor building purge filtration system.

Objective

To assure that the reactor building purge filtration system will perform within acceptable levels of efficiency and reliability.

Specification

3.22.1 The reactor building purge filtration system shall be operable whenever irradiated fuel handling operations are in progress in the reactor building and shall have the following performance capabilities:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows ($\pm$ 10%) on HEPA filters and charcoal adsorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal.

b. The results of laboratory carbon sample analysis shall show $\geq$ 90% radioactive methyl iodide removal at a velocity within $\pm$ 20% of system design, 0.05 to 0.15 mg/m^3 inlet methyl iodide concentration, $\geq$ 70% R. H. and $\geq$125F.

c. Fans shall be shown to operate within $\pm$ 10% design flow.

d. The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be less than 6 inches of water at system design flow rate ($\pm$ 10%).

e. Air distribution shall be uniform within $\pm$ 20% across HEPA filters and charcoal adsorbers when tested initially and after any maintenance or testing that could affect the air distribution within the reactor building purge filtration system.

3.22.2 If the requirements of Specification 3.22.1 cannot be met, either:

a. Irradiated fuel movement shall not be started (any irradiated fuel assembly movement in progress may be completed); or

b. Isolate the reactor building purge system.

3.22.3 The provisions of Specifications 3.0.3 and 3.0.4 are not applicable. |

Bases

The reactor building purge filtration system is designed to filter the reactor building atmosphere during normal operations for ease of personnel entry into the reactor building. This specification is intended to require the system operable during fuel handling operations, if the system

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 57 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 November 28, 1980, 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, which would change the Technical Specifications (TSs) to redefine the term OPERABLE and add general Limiting Conditions for Operation (LCOs).

Background

By letter dated April 10, 1980, the NRC requested all power reactor licensees to submit proposed TSs related to preserving the single failure criterion for systems that are relied upon in the licensees' Final Safety Analysis Report (FSAR). By and large, the single failure criterion is preserved in the TSs by properly defining the term OPERABLE and by specifying LCOs that require all redundant components of safety related systems to be OPERABLE. Our April 10 letter transmitted Model TSs for use by the licensees.

Evaluation

The NRC Model TSs consisted of three sections: (1) a definition of Operable, (2) an LCO providing an Action Statement for circumstances in excess of those addressed in an existing plant TS, and (3) an LCO concerning unavailability of emergency power or normal power. The licensee proposed a general section to the Limiting Conditions for Operation section of the TSs which included the NRC Model TSs and the associated TSs similar to the NRC Standard Technical Specifications (STS) (NUREG-0103, Revision 4). Many of the current TSs defining LCOs provide for no action statement if the LCO is not met. The licensee's proposed TS changes would apply separate action statements where the NRC Model TS action statement would not apply and would apply an exception to the NRC Model TSs where no action statement is applicable. These proposed changes were made in an acceptable manner.

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The NRC staff requested the licensee to revise the definition of Operable to implicitly state that a system is capable of performing its specified function when all necessary instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system to perform its function are also capable of performing their related support function. We have reviewed the licensee's definition and have determined that it is consistent with our request and is therefore acceptable.

After our letter of April 10, 1980, and the licensee's application dated Nobember 28, 1980, we have determined that the time allowed to come to hot standby were too restrictive and could, if problems arise during shutdown, lead to a reactor trip. To avoid unnecessary reactor trips, we have proposed to lengthen the times to come to hot standby and to modify the TS accordingly. The licensee agreed to this modification.

The licensee's proposed TSs, as modified for the LCOs, for the most part, are similar or identical to the NRC Model TSs and the NRC STS. We have discussed with the licensee's staff the areas of deviation and omission from the Model TSs. The licensee's staff agreed to modifications to the proposed TSs which would be consistent with the intent of the Model TSs. We, therefore, find the proposed TSs, as modified, acceptable.

We conclude that the redefinition of OPERABLE and the addition of LCOs, described above, will aid in preserving the safety related systems single failure criterion and are thus 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: June 16, 1981

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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. 57 to Facility Operating License No. DPR-51, issued to Arkansas Power & 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 its date of issuance.

The amendment modifies the ANO-1 Appendix A Technical Specifications by providing a redefinition of the term "Operable" and the addition of general Limiting Conditions for Operation.

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.

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For further details with respect to this action, see (1) the licensee's application dated November 28, 1980, (2) Amendment No. 57 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 16th day of June 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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