Docket File
DCS MS-016

MAR 4 1982

Docket No. 50-368

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



Dear Mr. Cavanaugh:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 29 to Facility Operating License No. NPF-6 for the Arkansas Power & Light Company for the Arkansas Nuclear One Unit 2 plant. The amendment consists of changes to the list of conditions in the body of the license and of changes to the Technical Specifications (TS) as listed below.

1. Reactor Building/Containment Cooler TS 3.6.2.3 in response to an application dated September 22, 1981.

2. Health Physics Section Organization TS in response to an application dated October 8, 1981.

3. Main feedwater isolation valve license condition 2.C.3.L.

4. Emergency feedwater pump response time TS Table 3.3-5 in response to an application dated July 24, 1979 as supplemented August 1, 1980 and May 6, 1981.

5. Decay heat removal capability TS 3/4 4.1 and TS 3.9.8 in response to an application dated October 31, 1980.

6. Secondary Coolant Activity TS 3.7.1.4 in response to an application dated March 5, 1981 as supplemented on May 6, 1981.

7. Miscellaneous Corrections to TS Bases 3/4.3.3.6 associated with the issuance of Amendments Nos. 20 and 22 and to TS 3.2.8 associated with the issuance of Amendment No. 24.



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During our review of your proposed amendments we found that certain modifications were necessary to meet our requirements. Your staff has agreed to these modifications and the item 3 and 7 modifications and they have been incorporated in this amendment.

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

Sincerely,

Original signed by:

Robert E. Martin, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 29 to NPF-6
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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| SURNAME | PMKreutzer | RMartin/pn | RAClark | TMNovak | B. Jones | RWHouston | |
| DATE | 1/6/82 | 1/26/82 3/03/82 | 1/27/82 | 1/ /82 | 2/1/82 | 1/26/82 | |

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

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PMKreutzer

Docket No. 50-368

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: ARKANSAS POWER AND LIGHT COMPANY, Arkansas Nuclear One, Unit No. 2

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

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: ~~Amendment No. 29/~~
Referenced documents have been provided PDR.

Division of Licensing
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

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|---|---|---|---|---|---|---|
| OFFICE | ORB#3:DL | | | | | |
| SURNAME | PMKreutzer/pn | | | | | |
| DATE | 3/ 4/82 | | | | | |

NRC FORM 102 7—79

Arkansas Power & Light Company

cc:

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U.S. Environmental Protection Agency
Region VI Office
ATTN: Regional Radiation Representative
1201 Elm Street
Dallas, Texas 75270

cc w/enclosure(s) and incoming
dated: 9/22/81, 10/8/81, 7/24/79, 8/1/80, 5/6/81, 10/31/80, 3/5/81, 5/6/81
S. L. Smith, Operations Officer
Arkansas Nuclear Planning & Response Program
P. O. Box 1749
Russellville, Arkansas 72801

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

ARKANSAS POWER AND LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 29
License No. NPF-6

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

   A. The applications for amendment by Arkansas Power and Light Company (the licensee) dated September 22, 1981; October 8, 1981; July 24, 1979 as supplemented August 1, 1980 and May 6, 1981; October 31, 1980 and March 5, 1981 as supplemented May 6, 1981, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, by deletion of license condition 2.C.3.L, and by amending paragraph 2.C.(2) of Facility Operating License No. NPF-6 to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 29, 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the
Technical Specifications

Date of Issuance: March 4, 1982

## ATTACHMENT TO LICENSE AMENDMENT NO. 29

## FACILITY OPERATING LICENSE NO. NPF-6

## DOCKET NO. 50-368

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

Pages - Appendix A

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## LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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BASES

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## AXIAL SHAPE INDEX

### LIMITING CONDITION FOR OPERATION

3.2.7 The core average AXIAL SHAPE INDEX (ASI) shall be maintained within the following limits:

a. COLSS OPERABLE
$-0.28 \leq \text{ASI} \leq +0.28$

b. COLSS OUT OF SERVICE (CPC)
$-0.20 \leq \text{ASI} \leq +0.20$

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER*

ACTION:

With the core average AXIAL SHAPE INDEX (ASI) exceeding its limit, restore the ASI to within its limit within 2 hours or reduce THERMAL POWER to less than 20% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.6 The core average AXIAL SHAPE INDEX shall be determined to be within its limits at least once per 12 hours using the COLSS or any OPERABLE Core Protection Calculator channel.

*See Special Test Exception 3.10.2.

## PRESSURIZER PRESSURE

### LIMITING CONDITION FOR OPERATION

3.2.8 The average pressurizer pressure shall be maintained between 2225 psia and 2275 psia.

APPLICABILITY: MODE 1

ACTION:

With the average pressurizer pressure exceeding its limits, restore the pressure to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

### SURVEILLANCE REQUIREMENTS

4.2.6 The average pressurizer pressure shall be determined to be within its limit at least once per 12 hours.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 1. Manual | |
| a. SIAS | |
| Safety Injection | Not Applicable |
| b. CSAS | |
| Containment Spray | Not Applicable |
| c. CIAS | |
| Containment Isolation | Not Applicable |
| d. MSIS | |
| Main Steam Isolation | Not Applicable |
| e. CCAS | |
| Containment Cooling | Not Applicable |
| f. RAS | |
| Containment Sump Recirculation | Not Applicable |
| g. EFAS | |
| Train A | Not Applicable |
| Train B | Not Applicable |
| 2. Pressurizer Pressure-Low | |
| a. Safety Injection | |
| 1) High Pressure Safety Injection | $\leq$ 30* |
| 2) Low Pressure Safety Injection | $\leq$ 35* |
| 3. Containment Pressure-High | |
| a. Safety Injection | |
| 1) High Pressure Safety Injection | $\leq$ 31.6* |
| 2) Low Pressure Safety Injection | $\leq$ 51.6* |
| b. Containment Isolation | $\leq$ 52.1*/37.1** |
| c. Containment Cooling | $\leq$ 43.1*/28.1** |

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

| INITIATING SIGNAL AND FUNCTION | RESPONSE TIME IN SECONDS |
|---|---|
| 4. Containment Pressure--High-High | |
| a. Containment Spray | ≤ 42.1*/27.1** |
| 5. Steam Generator Pressure-Low | |
| a. Main Steam Isolation | ≤ 3.9 |
| b. Feedwater Isolation | ≤ 36.4*/21.4** |
| 6. Refueling Water Tank-Low | |
| a. Containment Sump Valve Open | ≤ 145.0 |
| 7. Steam Generator Level-Low | |
| a. Emergency Feedwater - Train A | ≤ 97.4 |
| b. Emergency Feedwater - Train B | ≤ 112.4*/97.4** |
| 8. Steam Generator ΔP-High Coincident With Steam Generator Level-Low | |
| a. Emergency Feedwater - Train A | ≤ 97.4 |
| b. Emergency Feedwater - Train B | ≤ 112.4*/97.4** |

TABLE NOTATION

*Diesel generator starting and sequence loading delays included.

**Diesel generator starting delays not included, sequence loading delays included. Offsite power available.

## 3/4.4 REACTOR COOLANT SYSTEM

### 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

### STARTUP AND POWER OPERATION

### LIMITING CONDITION FOR OPERATION

3.4.1.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

With less than the above required reactor coolant pumps in operation, be in at least HOT STANDBY within 1 hour.

### SURVEILLANCE REQUIREMENTS

4.4.1.1 The above required reactor coolant loops shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

* See Special Test Exception 3.10.3.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be operable:

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

b. At least one of the above Reactor Coolant Loops shall be in operation.*

APPLICABILITY: MODE 3.

ACTION:

a. With less than the above required reactor coolant loops operable, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 At least the above required reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.2 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.

3. Shutdown Cooling Loop (A)#.

4. Shutdown Cooling Loop (B)#.

b. At least one of the above coolant loops shall be in operation.*

APPLICABILITY: Modes 4 and 5.

ACTION:

a. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours.

b. With no coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required shutdown cooling loop(s) shall be determined OPERABLE per Specification 4.0.5.

4.4.1.3.2 The required reactor coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 23% indicated level at least once per 12 hours.

4.4.1.3.4 At least one coolant loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

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* All reactor coolant pumps and decay heat removal pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

# The normal or emergency power source may be inoperable in Mode 5.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a CSAS test signal.

   2. Verifying that each sodium hydroxide addition pump starts automatically on a CSAS test signal.

e. At least once per 5 years by verifying the flow rate through each component and pipe section in each sodium hydroxide injection path from the tank to the containment spray pump discharge piping to be at least 14 gpm.

## LIMITING CONDITION FOR OPERATION

3.6.2.3 Two independent containment cooling groups shall be OPERABLE with at least one operational cooling unit in each group.

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

ACTION:

a. With one group of the above required containment cooling units inoperable and both containment spray systems OPERABLE, restore the inoperable group of cooling units to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two groups of the above required containment cooling units inoperable and both containment spray systems OPERABLE, restore at least one group of cooling units to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both above required groups of cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one group of the above required containment cooling units inoperable and one containment spray system inoperable, restore the inoperable spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the inoperable group of containment cooling units to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4.6.2.3 Each containment cooling group shall be demonstrated OPERABLE:

a. At least once per 14 days by:

1. Verifying a service water flow rate of ≥ 1250 gpm to each group of cooling units; each unit within the group having an operable fan, or by verifying a service water flow rate of ≥ 1250 gpm to one unit within the group; that unit having an operable fan.

2. Chlorinating the service water during the surveillance in 4.6.2.3.a.1 above, whenever service water temperature is between 60°F and 80°F.

b. At least once per 31 days by:

1. Starting (unless already operating) each operational cooling unit from the control room.

2. Verifying that each operational cooling unit operates for at least 15 minutes.

c. At least once per 18 months by verifying that each cooling unit starts automatically on a CCAS test signal.

## SHUTDOWN COOLING AND COOLANT CIRCULATION

## SHUTDOWN COOLING - ONE LOOP

### LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one shutdown cooling loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The shutdown cooling loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS.

c. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.8.1 A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 3000 gpm at least once per 24 hours.

## SHUTDOWN COOLING - TWO LOOPS

### LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent shutdown cooling loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, immediately initiate corrective action to return the loops to OPERABLE status as soon as possible.

b. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.8.2 The required shutdown cooling loops shall be determined OPERABLE per Specification 4.0.5.

* The normal or emergency power source may be inoperable for each shutdown cooling loop.

## WATER LEVEL - REACTOR VESSEL

### LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

## 3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

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

## 3/4.3.3.7 CHLORINE DETECTION SYSTEMS

The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.

## 3/4.3.3.8 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate 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 and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, except for detectors located in the containment during Modes 1 and 2, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

## 3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

## 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above the limits specified by TS 3.2.4 during all normal operations and anticipated transients.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

## 3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

## 3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The steam bubble functions to relieve RCS pressure during all design transients.

The requirement that 150 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss-of-offsite power condition to maintain natural circulation at HOT STANDBY.

## 3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 0.5 GPM per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

## BASES

### 3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses.

### 3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

### 3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

### 3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY of the containment purge and exhaust system HEPA filters and charcoal adsorbers ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Operation of the containment purge and exhaust system HEPA filters and charcoal adsorbers and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

## 3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

## 3/4.9.6 REFUELING MACHINE OPERABILITY

The OPERABILITY requirements for the refueling machine ensure that: 1) the refueling machine will be used for movement of CEAs with fuel assemblies and that it has sufficient load capacity to lift a fuel assembly, and 2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

## 3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

## 3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling 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 below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the core ensures that a single failure of the operating shutdown cooling 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 shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

# 6.0 ADMINISTRATIVE CONTROLS

## 6.1 RESPONSIBILITY

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

## 6.2 ORGANIZATION

### OFFSITE

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

### FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

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

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

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

e. All CORE ALTERATIONS shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

f. A site Fire Brigade of at least 5 members shall be maintained onsite at all times. The Fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE

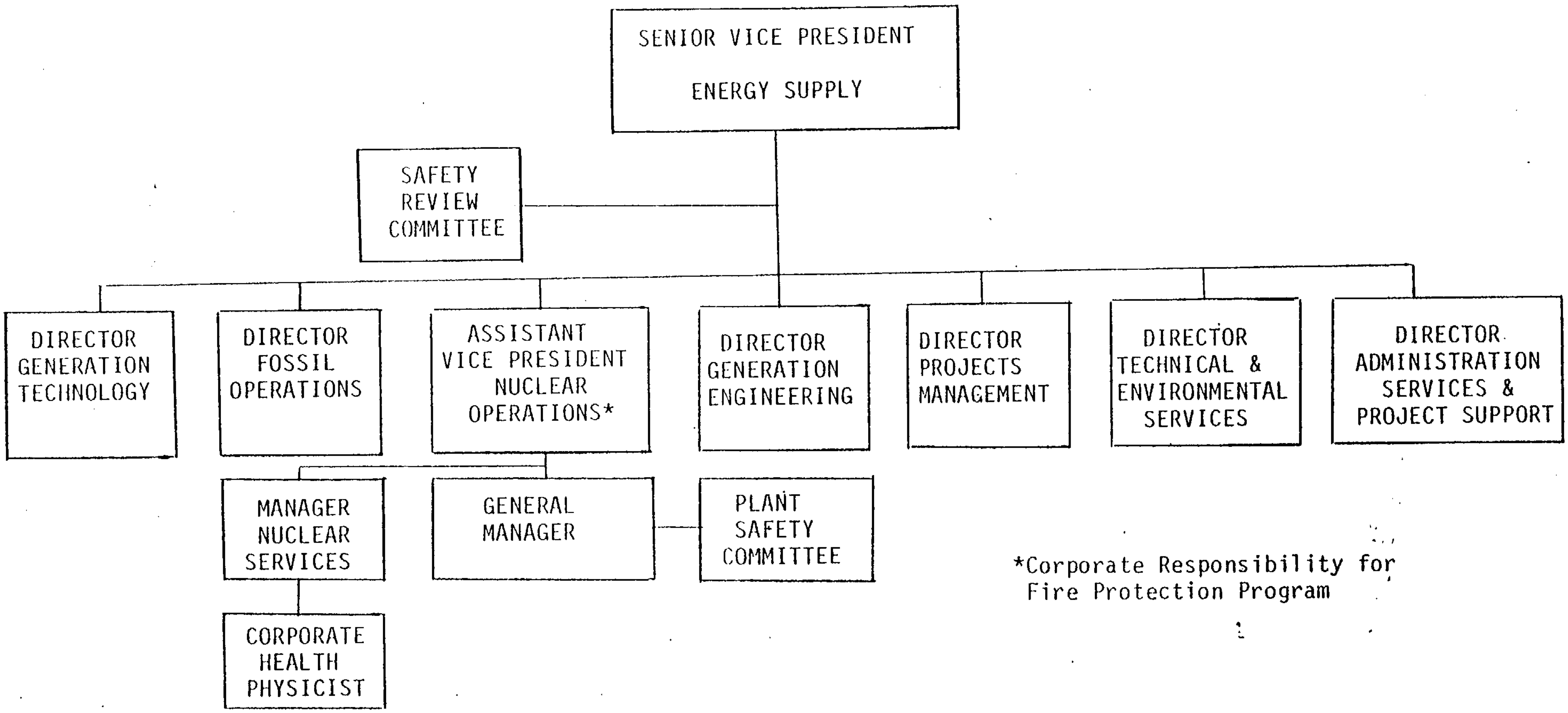
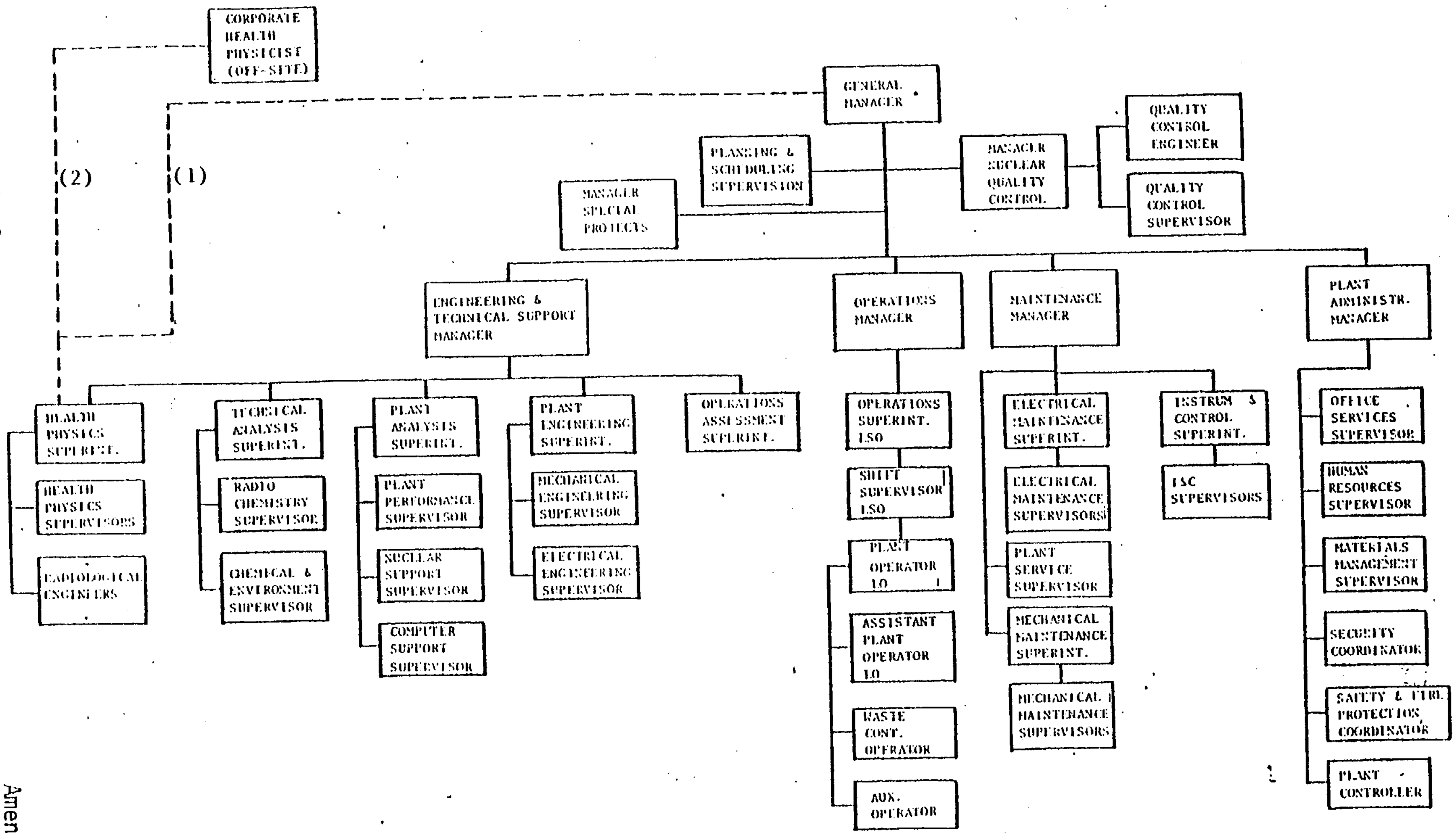



Figure 6.2-1 Management Organization Chart

ARKANSAS POWER AND LIGHT COMPANY
ARKANSAS NUCLEAR ONE




CODE: LSO-SENIOR OPERATOR LICENSE REQUIRED
LO - OPERATOR LICENSE REQUIRED
*ONSITE RESPONSIBILITY FOR FIRE PROTECTION PROGRAM

(1) The Health Physics Superintendent reports to the Manager, Engineering and Technical Support in administrative matters and routine health physcis concerns and he reports to the General Manager in matters of radiological health, safety and policy.

(2) The Health Physics Superintendent has direct interface with the Corporate Health Physicist in matters of radiological health and safety. The Corporate Health Physicist reports to the Manager, Nuclear Services. He will help formulate Corporate Health Physics Policy and ensure that it is properly implemented.

FIGURE 6.2-2 Functional Organization for Plant Operation

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

| LICENSE CATEGORY | APPLICABLE MODES | |
|---|---|---|
| | 1, 2, 3 & 4 | 5 & 6 |
| SOL | 1 | 1* |
| OL | 2 | 1 |
| Non-Licensed | 2 | 1 |
| Shift Technical Advisor | 1 | None Required |

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS.

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

## 6.3 UNIT STAFF QUALIFICATIONS

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

## 6.4 TRAINING

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

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the General Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code - 1975, except for Fire Brigade training sessions which shall be held at least quarterly.

## 6.5 REVIEW AND AUDIT

### 6.5.1 PLANT SAFETY COMMITTEE (PSC)

FUNCTION

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

COMPOSITION

6.5.1.2 The Plant Safety Committee shall be composed of the:

| | |
|---|---|
| Chairman: | Manager of Special Projects |
| Member: | Operations Manager |
| Member: | Maintenance Manager |
| Member: | Engineering and Technical Support Manager |
| Member: | Administrative Manager |
| Member: | Technical Analysis Superintendent |
| Member: | Plant Analysis Superintendent |
| Member: | Plant Engineering Superintendent |
| Member: | Health Physics Superintendent |
| Member: | Nuclear Software Expert* |

The General Manager shall designate in writing the Alternate Chairman in the absence of the PSC Chairman.

* See page 6-5a

*If one of the above members of the Plant Safety Committee meets the qualification requirements for this position, the requirement to have this member is satisfied. This membership may be filled by two appropriately qualified individuals who shall ballot with a single combined vote. Generic qualifications for this membership shall be as follows:

One Individual

The Nuclear Software Expert shall have as a minimum a Bachelor's degree in Science or Engineering, Nuclear preferred (in accordance with ANSI N18.1). In addition, he shall have a minimum of four years of technical experience, of which a minimum of two years shall be in Nuclear Engineering and a minimum of two years shall be in Software Engineering. (Software Engineering is that branch of science and technology which deals with the design and use of software. Software Engineering is a discipline directed to the production and modification of computer programs that are correct, efficient, flexible, maintainable, and understandable, in reasonable time spans, and at reasonable costs). The two years of technical experience in Software Engineering may be general software experience not necessarily related to the software of the Core Protection Calculator System. One of these two years of experience shall be with certified computer programs.

Two Individuals

One of the individuals shall meet the requirements of the Nuclear Engineering portion of the above. The second individual shall have a Bachelor of Science degree (digital computer speciality) and meet the Software Engineering requirements of the above.

The membership (the Nuclear Software Expert or the Digital Computer Specialist) shall be knowledgeable of the Core Protection Calculator System with regard to:

a. The software modules, their interactions with each other and with the data base.

b. The relationship between operator's module inputs and the trip variables.

c. The relationship between sensor input signals and the trip variable.

d. The design basis of the Core Protection Calculator System.

e. The approved software change procedure and documentation requirements of a software change.

f. The security of the computer memory and access procedures to the memory.

6.12.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

## 6.13 HIGH RADIATION AREA

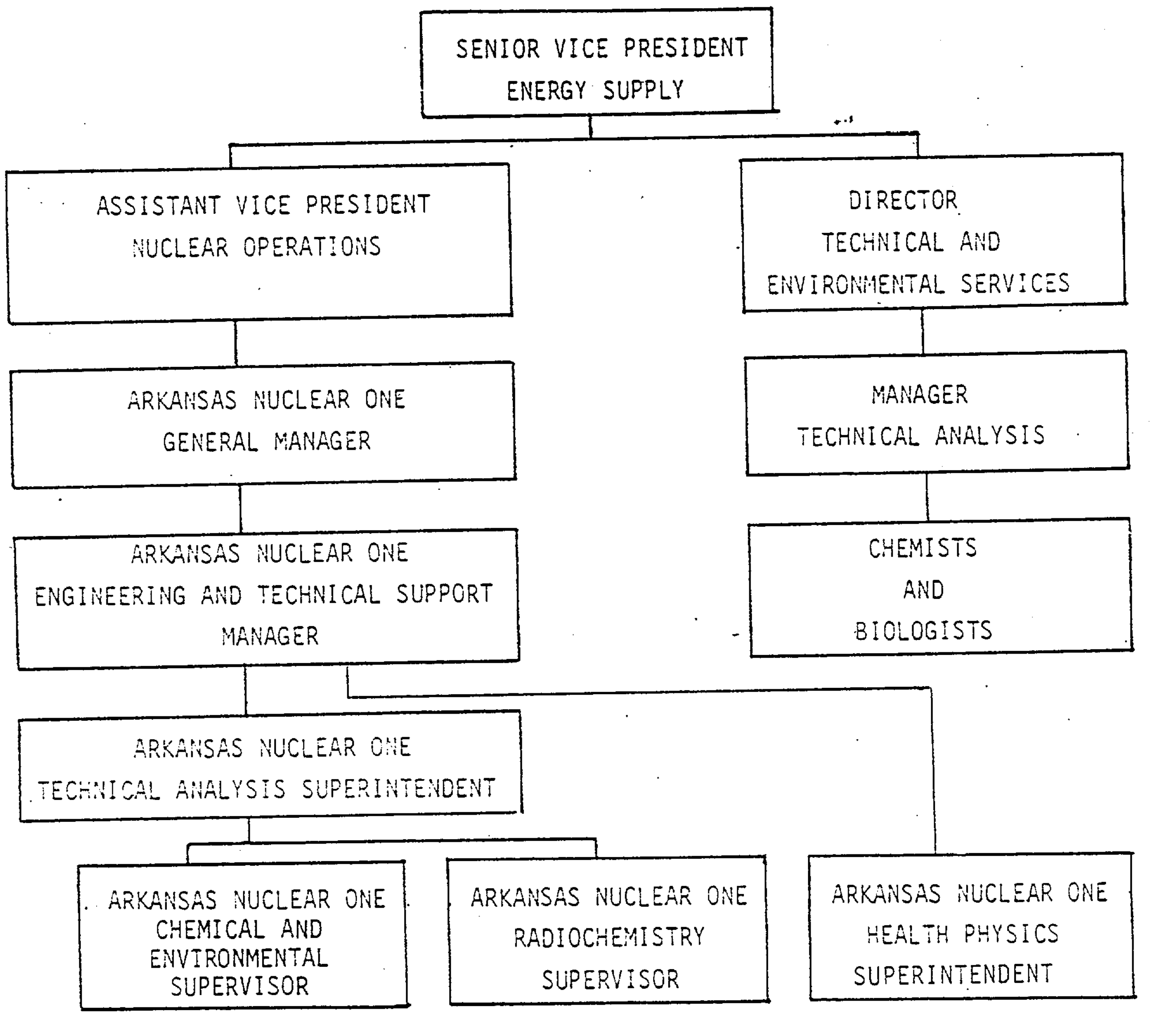
6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area (as defined in 20.202(b)(3) of 10 CFR 20) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring the issuance of a radiation work permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a present integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation work permit.

6.13.2 The requirements of 6.13.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and access to these areas shall be maintained under the administrative control of the Shift Supervisor on duty and/or the Health Physics Superintendent.




| | | |
|---|---|---|
| ARKANSAS POWER & LIGHT CO.<br>ARKANSAS NUCLEAR ONE | ENVIRONMENTAL SURVEILLANCE<br>ORGANIZATION CHART | FIGURE NO.<br>5-1 |

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 29 TO FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

Introduction

This Safety Evaluation addresses a number of changes to the body of the license and to the Technical Specifications (TS) as discussed in each of the following sections.

I. Reactor Building/Containment Cooling Unit Surveillance

By application dated September 22, 1981 the licensee proposed changes to the ANO-2 Technical Specifications (TS) for the containment cooling systems. The proposed change would remove an inconsistency between TS 3.6.2.3.a and 3.6.2.3.d.

TS 3.6.2.3.a specifies the required actions when one group of containment coolers is inoperable. There is a total of four containment cooling units (CCUs) with 2 CCUs per containment cooling group. TS 3.6.2.3.d addresses the circumstances wherein one CCU in a group may be inoperable; however, provided the remaining CCU in that group can be verified to pass the required amount of service water coolant to meet the assumptions of the safety analyses that group of containment coolers may continue to be declared operable. The staff has previously evaluated and found acceptable the situation wherein one CCU per group is OPERABLE in the evaluation supporting the issuance of Amendment No. 16, issued October 9, 1980.

Since TS 3.6.2.3.a and d are comparable in that they each refer to the action required if one group is inoperable the licensee proposes that the time allowed to return the inoperable group to an OPERABLE status should be the same in both specifications. We agree and would concur with the licensee's proposal to change the time allowed before further action must be taken from 72 hours to 7 days in TS 3.6.2.3.d. TS 3.6.2.3.d would then be consistent with TS 3.6.2.3.a and with the recommendations of the Standard TS.

However, the staff notes that TS 3.6.2.3.d is specific to the ANO plants and with the several modifications made to it in Amendment No. 16 and as discussed above, may be confusing. Therefore the staff believes that the licensee's proposed changes of September 22, 1981 should be modified to remove TS 3.6.2.3.d and put the same requirements into surveillance

TS 4.6.2.3.a.1. This is appropriate for the following reasons: (1) The resulting TS will tend to be less confusing, (2) the flowrate requirements (> 1250 gpm, etc.) properly belong in the surveillance TS with the other flowrate surveillance specifications, and (3) TS 3.6.2.3 will be consistent with the STS which will provide for improved standardization of the ANO-2 TS with the current STS. These changes have been discussed with and concurred in by the licensee. On these bases these changes are acceptable.

II. Reorganization of Health Physics Section

By application dated October 8, 1981 the licensee proposed to amend the portion of the TS addressing the organization of the plant staff health physics section. The changes consist of creation of the position of Health Physics Superintendant and the addition of a communication interface with the Corporate Health Physicist who is located offsite.

The licensee states that this revision of the ANO organization addresses a significant finding of the NRC's Radiological Appraisal Team during October/November, 1980 and also fulfills a commitment made in an AP&L Co. letter dated April 16, 1981 to the Office of Inspection and Enforcement. We have reviewed the proposed changes and agree with the licensee's statement that they will improve the ANO Radiation Protection Program by elevating the Health Physics Section of the plant staff to a level comparable with the Operations and the Maintenance Groups and will ensure that the Health Physics Superintendant has direct access to responsible management in matters of radiological health, safety and policy. We have determined that the proposed changes are in agreement with the commitments made in the licensee's April 16, 1981 letter and are acceptable.

III. Main Feedwater Isolation Valves

License condition 2.C.3.h requires the licensee to install one additional main feedwater isolation valve in each of the main feedwater lines prior to startup following the first regularly scheduled refueling outage. The required modifications are described in the Final Safety Analysis Report Section 6.2.1.1.2.6. The staff's evaluation of the technical issue leading to the requirement to install these valves has previously been presented in Supplement No. 2 to the staff's Safety Evaluation Report NUREG-0308 issued in September 1978. The staff has verified, through an inspection conducted by the Office of Inspection and Enforcement, that the valves have been installed as required by the license condition. Therefore, we conclude that the terms of the license condition have been satisfied and the condition may be deleted from the license.

IV. Emergency Feedwater Pump Response Time

By application dated July 24, 1979 the licensee proposed to change TS Table 3.3-5 items 7a and 8a from ≤ 21.4 seconds to 97.4 seconds. Items 7a and 8a currently specify the required response time for the steam driven emergency feedwater (EFW) pump as 21.4 seconds. Items 7b and 8b specify the required response time for the motor driven EFW pump as 97.4 seconds when offsite power is available. The 97.4 seconds for the motor driven pumps was chosen to ensure that the provision of emergency feedwater to the steam generators will occur within the time limit assumed in the accident analysis. The licensee states that no credit was taken for delivery of EFW from either pump prior to 97.4 seconds following an actuation signal.

We have reviewed the licensee's request and conclude that, based on a required response time of 97.4 seconds to meet the assumptions of the accident analyses, the specification of a lower response time for the steam driven pump is unnecessary and may be modified to a value of 97.4 seconds.

V. Decay Heat Removal Capability

By application dated October 31, 1980 the licensee proposed to amend the ANO-2 TS in response to the staff's letter to all PWR licensees dated June 11, 1980. The proposed changes are directed at ensuring that redundancy in decay heat removal capability will be maintained in all operating modes. The staff's acceptance of the requested changes is contained within the June 11, 1980 letter which also constitutes the staff's safety evaluation of this matter. We have reviewed the licensee's application and have determined that it is consistent with the decay heat removal capability guidance presented in the staff's June 11, 1980 letter. We find the licensee's application to be acceptable.

VI. Secondary Coolant Activity

By application dated March 5, 1981 the licensee proposed to reduce the maximum secondary coolant activity allowed by TS 3.7.1.4 from 0.10 to 0.046 µCi/gram dose equivalent I-131 and the primary/secondary leak rate from 1.0 gpm to 100.gpday. The staff issued a request for additional information which was responded to by letter dated May 6, 1981. The licensee's basis for the change is that the values, 0.10 µCi/gm and 1.0 gpm, are the standard values contained in the Standard TS whereas the licensee's specific safety analyses for the ANO-2 plant are based on values of 0.046 µCi/gm and 100.gpd. Therefore the licensee proposes to change the TS to provide consistency between the LCOs in the TS and the assumptions made for the licensee's ANO-2 safety analyses.

The results of staff's previous evaluations of the main steamline break and the steam generator tube rupture are presented in Section 15.0 of the SER, NUREG-0308 issued November 1977. These evaluations were based in part on independent determinations by the staff of the event's radiological consequences utilizing the assumptions listed in SER Table 15.2 which include a secondary coolant activity of 0.10 µCi/gm and a primary/secondary leakrate of 1.0 gpm. In the SER the staff concluded that the consequences of these events could be acceptably controlled by limiting activity and leakrate to these values in the Technical Specifications. The staff reaffirms its findings made in the SER and concludes that on the basis of its independent determination the ANO-2 Technical Specifications do not require revision in order to ensure acceptable results for the main steamline break and steam generator tube break events.

VII. Miscellaneous Corrections

1. Amendment No. 20 to the license issued on March 23, 1981 made changes to TS BASES 3/4.3.3.6 which were deleted by mistake with the issuance of Amendment No. 22. Therefore, those Amendment 20 changes are hereby restored with the issuance of this amendment.

2. Amendment No. 24 to the license issued June 19, 1981 modified TS 3.2.8. The modified specification is an LCO on pressurizer pressure but was issued with its ACTION statement referring to temperature, not pressure. Therefore, with the issuance of this amendment the ACTION statement is corrected to refer to pressure.

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.

Date: March 4, 1982

Principal contributors to this SER were W. Pasedag and R. Martin.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-368

ARKANSAS POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 29 to Facility Operating License No. NPF-6 issued to Arkansas Power and Light Company (the licensee), which revised the Technical Specifications for operation of Arkansas Nuclear One, Unit No. 2, located in Pope County, Arkansas. The amendment is effective as of its date of issuance.

The amendment modifies the body of the license and the ANO-2 Appendix A and B Technical Specifications dealing with the reactor building/containment cooler surveillance requirements, the health physics section organization, the main feedwater isolation valve, the emergency feedwater pump response time, decay heat removal capability, secondary coolant activity and several miscellaneous corrections.

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

The Commission has determined that the issuance of the 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 the amendment.

For futher details with respect to this action, see (1) the licensee's applications dated September 22, 1981; October 8, 1981; July 24, 1979 as supplemented August 1, 1980 and May 6, 1981; October 31, 1980; and March 5, 1981 as supplemented May 6, 1981, (2) Amendment No. 29 to License No. NPF-6, and (3) the Commission's related Safety Evalution. 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 4th day of March, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing