



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. 76
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 5, 1982, as supplemented February 17, 1983, and April 7, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public;
and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

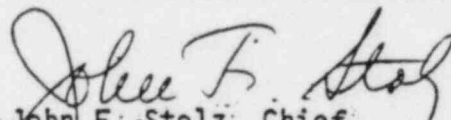
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76, 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, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 15, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 76

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 areas of change.

Page

v
59
59a
59b
59c (new page)
59d (new page)
116
127

3.5.2-2E	ROD POSITION LIMITS FOR TWO-PUMP OPERATION FROM 0 TO 60 EFPD-ANO-1, CYCLE 5	48c4
3.5.2-2F	ROD POSITION LIMITS FOR TWO-PUMP OPERATION FROM 50 TO 200 \pm 10 EFPD-ANO-1, CYCLE 5	48c5
3.5.2-2G	ROD POSITION LIMITS FOR TWO-PUMP OPERATION FROM 200 \pm 10 TO 400 \pm 10 EFPD-ANO-1, CYCLE 5	48c6
3.5.2-2H	ROD POSITION LIMITS FOR TWO-PUMP OPERATION FROM 400 \pm 10 TO 435 \pm 10 EFPD-ANO-1, CYCLE 5	48c7
3.5.2-3A	OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 TO 60 EFPD-ANO-1, CYCLE 5	48d
3.5.2-3B	OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 50 TO 200 \pm 10 EFPD-ANO-1, CYCLE 5	48d1
3.5.2-3C	OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 200 \pm 10 TO 400 \pm 10 EFPD-ANO-1, CYCLE 5	48d2
3.5.2-3D	OPERATIONAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 400 \pm 10 TO 435 \pm 10 EFPD-ANO-1, CYCLE 5	48d3
3.5.2-4	LOCA LIMITED MAXIMUM ALLOWABLE LINEAR HEAT RATE	48e
5.2-4A	ASPR POSITION LIMITS FOR OPERATION FROM 0 TO 60 EFPD-ANO-1, CYCLE 5	48f
3.5.2-4B	ASPR POSITION LIMITS FOR OPERATION FROM 50 TO 200 \pm 10 EFPD-ANO-1, CYCLE 5	48g
3.5.2-4C	ASPR POSITION LIMITS FOR OPERATION FROM 200 \pm 10 TO 400 \pm 10 EFPD-ANO-1, CYCLE 5	48h
3.5.2-4C	ASPR POSITION LIMITS FOR OPERATION FROM 400 \pm 10 TO 435 \pm 10 EFPD-ANO-1, CYCLE 5	48i
3.5-4-1	INCORE INSTRUMENTATION SPECIFICATION AXIAL IMBALANCE INDICATION	53a
3.5-4-2	INCORE INSTRUMENTATION SPECIFICATION RADIAL FLUX TILT INDICATION	53b
3.5-4-3	INCORE INSTRUMENTATION SPECIFICATION	53c
3.8.1	SPENT FUEL POOL ARRANGEMENT UNIT NO. 1	59c
3.8.2	MINIMUM BURNUP vs. INITIAL ENRICHMENT FOR REGION 2 STORAGE	59d
6.2-1	MANAGEMENT ORGANIZATION CHART	119
6.2-2	FUNCTIONAL ORGANIZATION FOR PLANT OPERATION	120

- 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. In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool. 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 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.

- 3.8.15 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.
- 3.8.16 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.
- 3.8.17 Storage in Region 2 (as shown on Figure 3.8.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.
- 3.8.18 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million.

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 9.6 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) Although the refueling boron 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: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.⁽³⁾; and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 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 3.8.15 is completed.

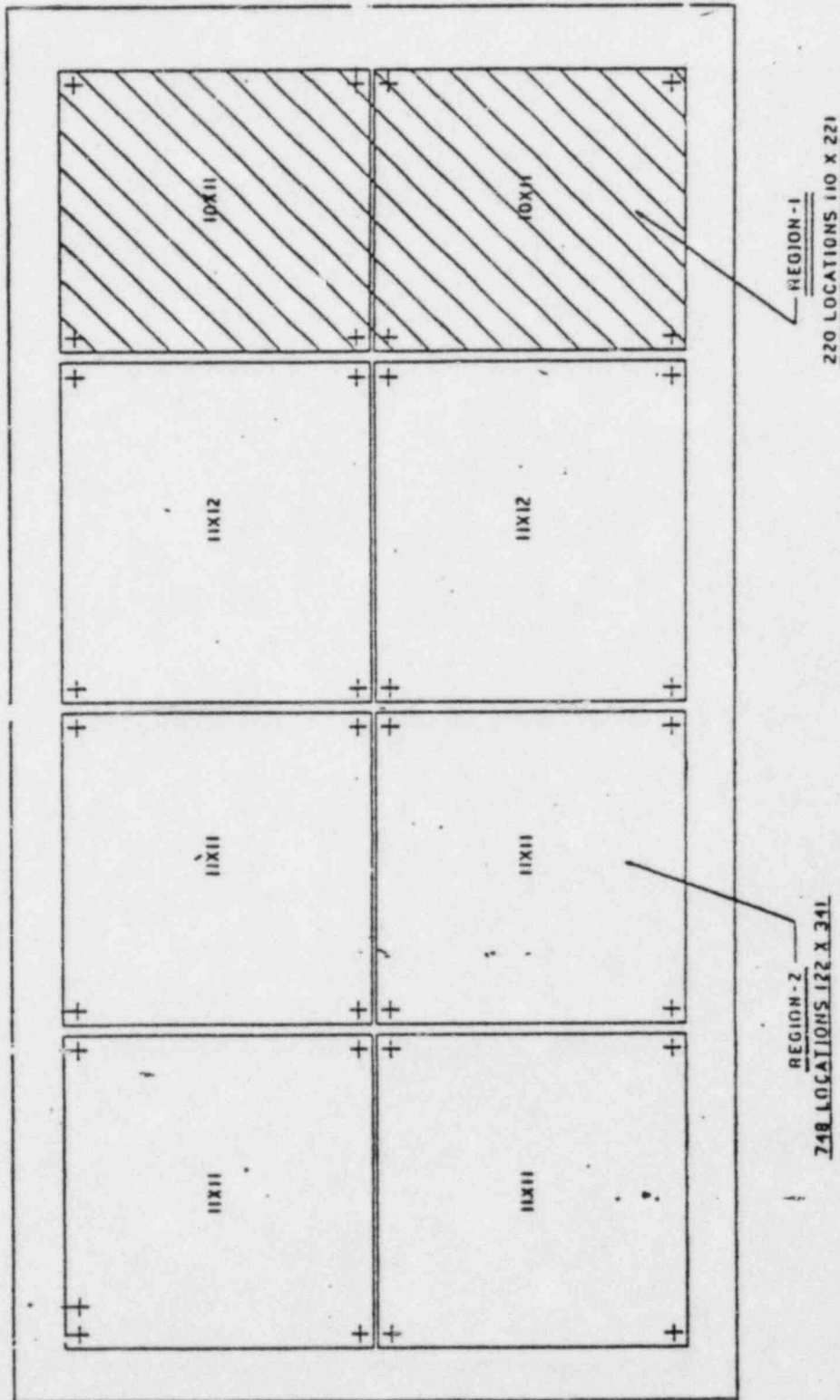
Specification 3.8.15 assures that the spent fuel cask drop accident cannot occur prior to completion of the NRC staff's review of this potential accident and the completion of any modifications that may be necessary to preclude the accident or mitigate the consequences. Upon satisfactory completion of the NRC's review, Specification 3.8.15 shall be deleted.

Specifications 3.8.16 and 3.8.17 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

Specification 3.8.18 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

REFERENCES

- (1) FSAR, Section 9.5
- (2) FSAR, Section 14.2.2.3
- (3) FSAR, Section 14.2.2.3.3

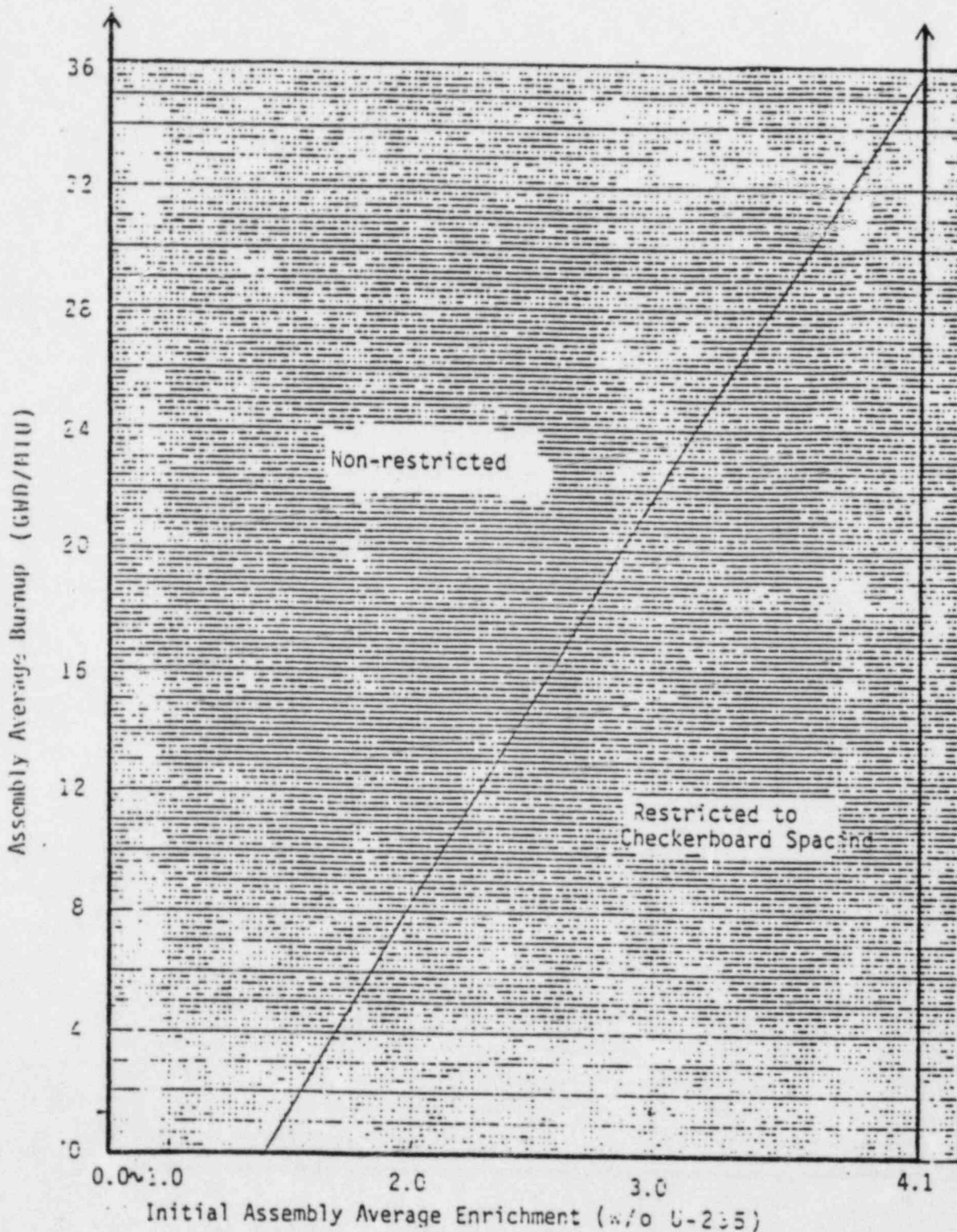


SPENT FUEL POOL ARRANGEMENT UNIT #1

Figure 3.8.1

FIGURE 3.8.2

MINIMUM BURNUP VS. INITIAL ENRICHMENT
FOR REGION 2 ENRICHMENT



5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Applicability

Applies to storage facilities for new and spent fuel assemblies.

Objective

To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.

Specification

5.4.1 New Fuel Storage

1. Fuel assemblies are stored in racks of parallel rows, having a nominal center to center distance of 21 inches in both directions. This spacing is sufficient to maintain a K_{eff} of less than .9 even if flooded with unborated water, based on fuel with an enrichment of 3.5 weight percent U235.
2. New fuel may be stored in the spent fuel pool or in its shipping containers.

5.4.2 Spent Fuel Storage

1. The spent fuel racks are designed and shall be maintained so that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) when the pool is flooded with unborated water.
2. The spent fuel pool and the new fuel pool racks are designed as seismic Class I equipment.

REFERENCES

FSAR, Section 9.6

- a. The facility shall be placed in at least hot shutdown within one hour.
- b. The Nuclear Regulatory Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.12.3.1.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

- a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.
- b. Refueling operations.
- c. Surveillance and test activities of safety related equipment.
- d. Security Plan implementation.
- e. Emergency Plan implementation.
- f. Fire Protection Program implementation.
- g. New and spent fuel storage.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

- a. The intent of the original procedure is not altered.
- b. The change is approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.
- c. The change is documented, reviewed by the PSC and approved by the General Manager within 14 days of implementation.



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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO.2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. NPF-5

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 5, 1982, as supplemented February 17, 1983, and April 7, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

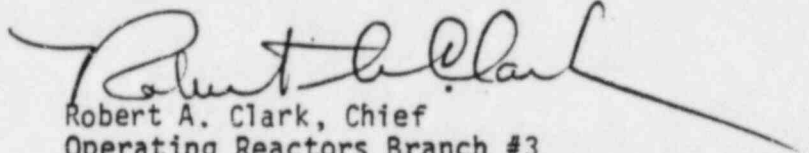
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-6 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 43, 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, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 15, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

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 areas of change. Corresponding overleaf pages are provided to maintain document completeness.

Pages

VIII

3/4 9-3

3/4 9-14

3/4 9-15

3/4 9-16

6-13

B 3/4 9-1

B 3/4 9-3

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.7 PLANT SYSTEMS</u>	
3/4.7.1 TURBINE CYCLE	
Safety Valves.....	3/4 7-1
Emergency Feedwater System.....	3/4 7-5
Condensate Storage Tank.....	3/4 7-7
Activity	3/4 7-8
Main Steam Isolation Valves	3/4 7-10
3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION.....	3/4 7-14
3/4.7.3 SERVICE WATER SYSTEM.....	3/4 7-15
3/4.7.4 EMERGENCY COOLING POND.....	3/4 7-16
3/4.7.5 FLOOD PROTECTION.....	3/4 7-16a
3/4.7.6 CONTROL ROOM EMERGENCY AIR CONDITIONING AND AIR FILTRATION SYSTEM.....	3/4 7-17
3/4.7.8 HYDRAULIC SHOCK SUPPRESSORS.....	3/4 7-22
3/4.7.9 SEALED SOURCE CONTAMINATION.....	3/4 7-27
3.4.7.10 FIRE SUPPRESSION SYSTEMS	
Fire Suppression Water System.....	3/4 7-29
Spray and/or Sprinkler Systems.....	3/4 7-33
Fire Hose Stations.....	3/4 7-35
3/4.7.11 PENETRATION FIRE BARRIERS.....	3/4 7-37
3/4.7.12 SPENT FUEL POOL STRUCTURAL INTEGRITY.....	3/4 7-38
<u>3/4.8 ELECTRICAL POWER SYSTEMS</u>	
3/4.8.1 A.C. SOURCES	
Operating.....	3/4 8-1
Shutdown.....	3/4 8-5

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
3/4.8.2 <u>ONSITE POWER DISTRIBUTION SYSTEMS</u>	
A. C. Distribution - Operating.....	3/4 8-6
A. C. Distribution - Shutdown	3/4 8-7
D. C. Distribution - Operating	3/4 8-8
D. C. Distribution - Shutdown	3/4 8-10
Containment Penetration Conductor Overcurrent Protective Devices	3/4 8-11
<u>3/4.9 REFUELING OPERATIONS</u>	
3/4.9.1 BORON CONCENTRATION	3/4 9-1
3/4.9.2 INSTRUMENTATION.....	3/4 9-2
3/4.9.3 DECAY TIME AND SPENT FUEL STORAGE.....	3/4 9-3
3/4.9.4 CONTAINMENT BUILDING PENETRATIONS.....	3/4 9-4
3/4.9.5 COMMUNICATIONS	3/4 9-6
3/4.9.6 REFUELING MACHINE OPERABILITY	3/4 9-7
3/4.9.7 CRANE TRAVEL - SPENT FUEL POOL BUILDING	3/4 9-8
3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION	3/4 9-9
3/4.9.9 WATER LEVEL - REACTOR VESSEL	3/4 9-10
3/4.9.10 SPENT FUEL POOL WATER LEVEL	3/4 9-11
3/4.9.11 FUEL HANDLING AREA VENTILATION SYSTEM	3/4 9-12
3/4.9.12 FUEL STORAGE	3/4 9-14
<u>3/4.10 SPECIAL TEST EXCEPTIONS</u>	
3/4.10.1 SHUTDOWN MARGIN	3/4 10-1
3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS	3/4 10-2
3/4.10.3 REACTOR COOLANT LOOPS	3/4 10-3
3/4.10.4 CENTER CEA MISALIGNMENT	3/4 10-4
3/4.10.5 MINIMUM TEMPERATURE FOR CRITICALITY.....	3/4 10-5

REFUELING OPERATIONS

DECAY TIME AND SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.3.a The reactor shall be subcritical for at least 72 hours.

3.9.3.b In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for less than 175 hours, suspend all operations involving movement of more than 70 fuel assemblies from the reactor pressure vessel to the spent fuel pool. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3.a The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

4.9.3.b The reactor shall be determined to have been subcritical for at least 175 hours by verification of the date and time of subcriticality prior to movement of the 71st irradiated fuel assembly from the reactor pressure vessel to the spent fuel pool.

REFUELING OPERATIONS

CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

- a. The equipment door closed and held in place by a minimum of four bolts,
- b. A minimum of one door in each airlock is closed, and
- c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:
 1. Closed by an isolation valve, blind flange, or manual valve, or
 2. Exhausting through OPERABLE containment purge and exhaust system HEPA filters and charcoal adsorbers.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4.1 Each of the above required containment penetrations shall be determined to be in its above required condition within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment.

4.9.4.2 The containment purge and exhaust system shall be demonstrated OPERABLE at the following frequencies:

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

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying with 31 days after removal that laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.
3. Verifying a system flow rate of 39,700 cfm \pm 10% during system operation when tested in accordance with ANSI N510-1975.
 - b. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.
 - c. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 39,700 cfm \pm 10%.
 - d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove \geq 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 39,700 cfm \pm 10%.
 - e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove > 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 39,700 cfm \pm 10%.

REFUELING OPERATIONS

FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12.a Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specification 3.0.3 are not applicable.

3.9.12.b Storage in Region 2 (as shown on Figure 3.9.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.9.2. In the event a checkerboard storage configuration is deemed necessary for a portion of Region 2, vacant spaces adjacent to the faces of any fuel assembly which does not meet the Region 2 burnup criteria (Non-Restricted) shall be physically blocked before any such fuel assembly may be placed in Region 2. This will prevent inadvertent fuel assembly insertion into two adjacent storage locations. The provisions of Specification 3.0.3 are not applicable.

3.9.12.c The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million.

APPLICABILITY: During storage of fuel in the spent fuel pool.

ACTION:

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined a fuel assembly has been placed in the incorrect Region until such time as the correct storage location is determined. Move the assembly to its correct location before resumption of any other fuel movement.

Suspend all actions involving the movement of fuel in the spent fuel pool if it is determined the pool boron concentration is less than 1601 ppm, until such time as the boron concentration is increased to 1601 ppm or greater.

SURVEILLANCE REQUIREMENTS

4.9.12.a Verify all fuel assemblies to be placed in the spent fuel pool had an initial enrichment of less than or equal to 4.1 w/o U-235 by checking the assemblies design documentation.

4.9.12.b Verify all fuel assemblies to be placed in Region 2 of the spent fuel pool are within the enrichment and burnup limits of Figure 3.9.2 by checking the assemblies design and burnup documentation.

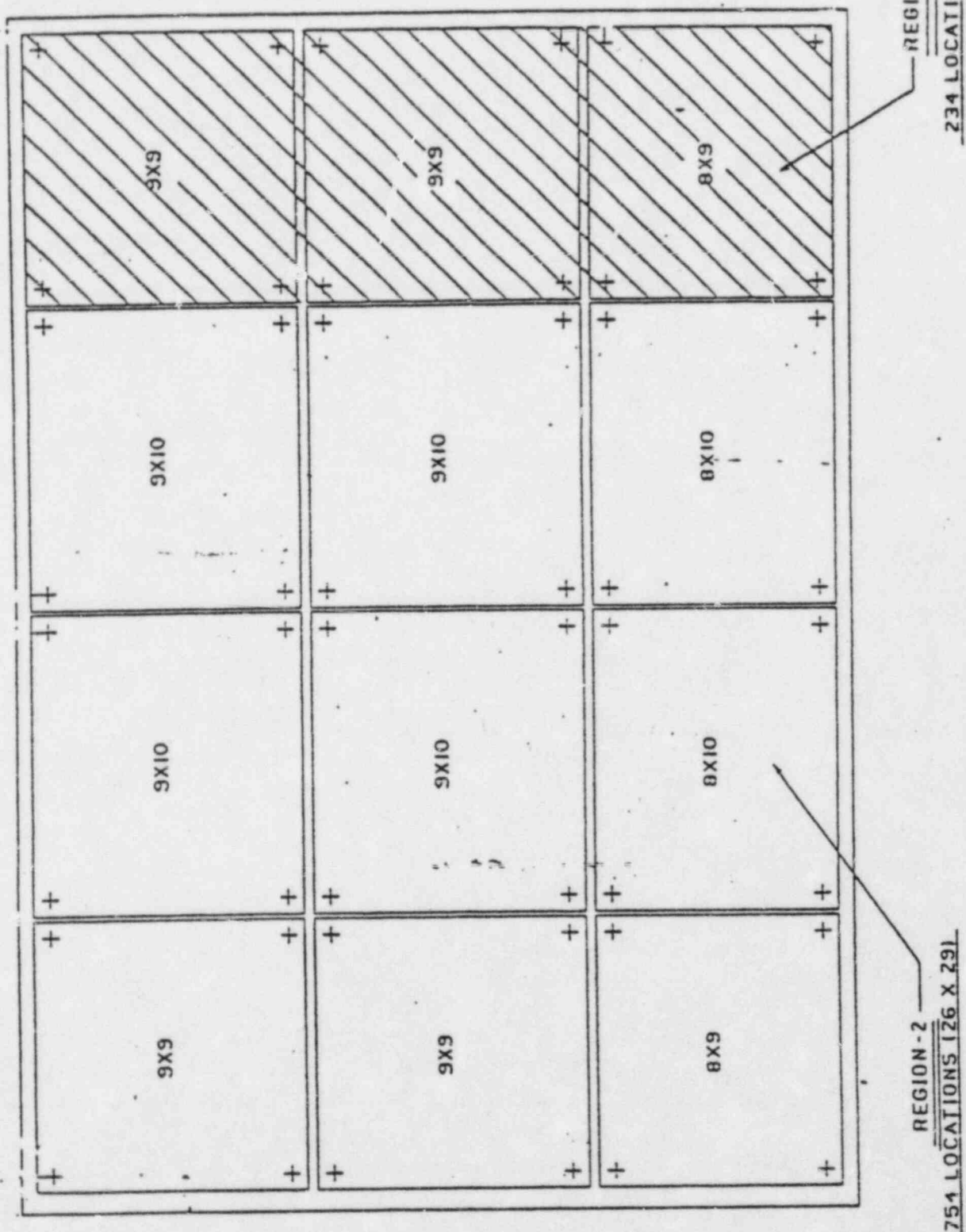
4.9.12.c Verify at least once per 31 days the spent fuel pool boron concentration is greater than 1600 ppm.



ARKANSAS-Unit 2

Amendment No. 43

3/4 9-15

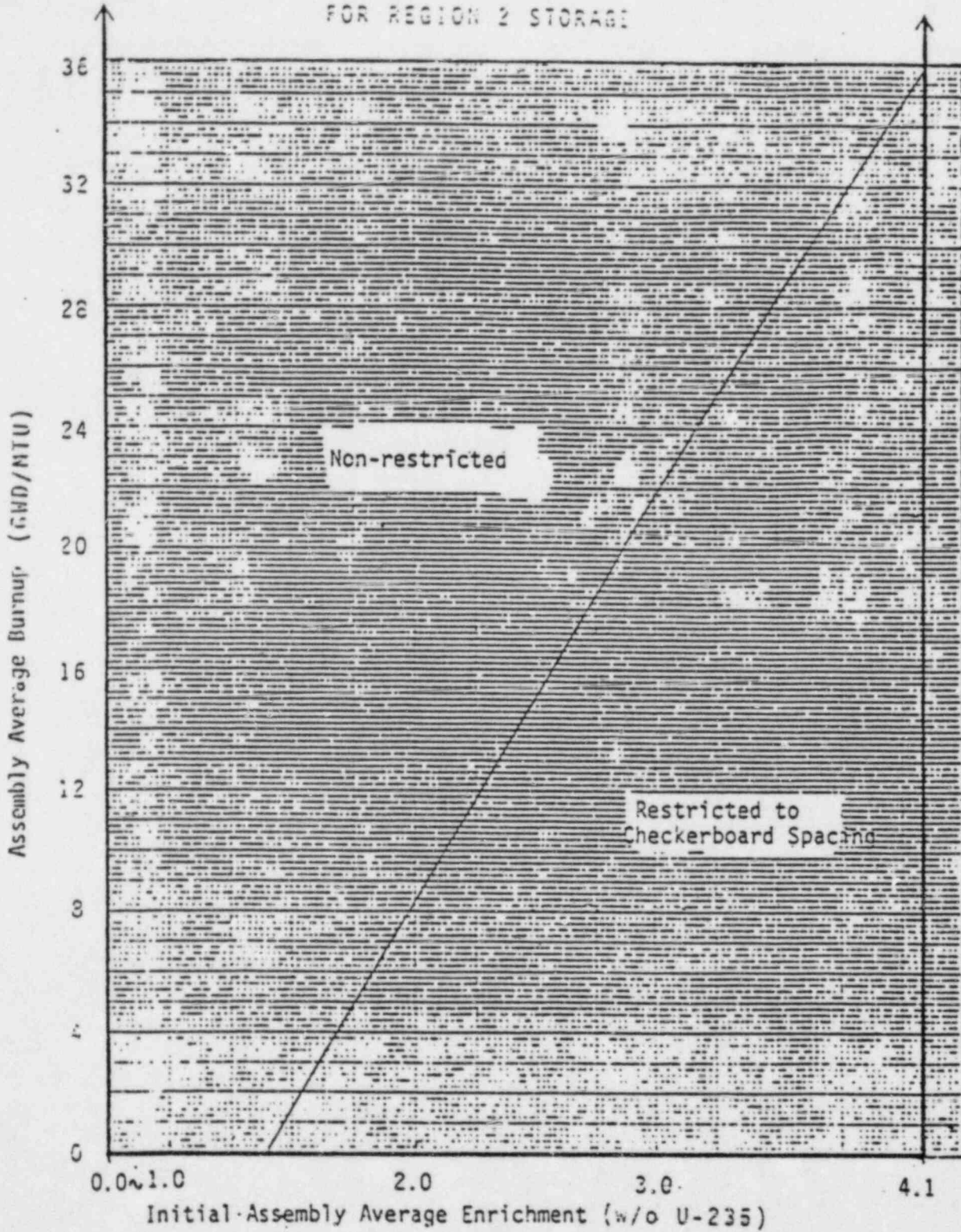


SPENT FUEL POOL ARRANGEMENT UNIT #2

Figure 3.9.1

FIGURE 3.9.2

MINIMUM BURMUP VS. INITIAL ENRICHMENT
FOR REGION 2 STORAGE



ADMINISTRATIVE CONTROLS

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

- a. The unit shall be placed in at least HOT STANDBY within one hour.
- b. The Safety Limit violation shall be reported to the Commission, the Vice President, Nuclear Operations and to the SRC within 24 hours.
- c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.
- d. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Vice-President, Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

- a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.
- b. Refueling operations.
- c. Surveillance and test activities of safety related equipment.
- d. Security Plan implementation.
- e. Emergency Plan implementation.
- f. Fire Protection Program implementation.
- g. Modification of Core Protection Calculator (CPC) Addressable Constants
NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the Plant Safety Committee.
- h. New and spent fuel storage.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

ADMINISTRATIVE CONTROLS

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

- a. The intent of the original procedure is not altered.
- b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.
- c. The change is documented, reviewed by the PSC and approved by the General Manager within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

3/4.9 REFUELING OPERATIONS

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.

The minimum requirement for reactor subcriticality prior to movement of more than 70 irradiated fuel assemblies to the spent fuel pool ensures that sufficient time has elapsed to allow radioactive decay of the short lived fission products such that the heat generated will not exceed the cooling capacity of the spent fuel pool cooling system. This decay time and total assembly limitation is conservatively within 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.

REFUELING OPERATIONS

BASES

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.

REFUELING OPERATIONS

BASES

3/4.9.9 and 3/4.9.10 WATER LEVEL-REACTOR VESSEL AND SPENT FUEL POOL WATER LEVEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gas activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 FUEL HANDLING AREA VENTILATION SYSTEM

The limitations on the fuel handling area ventilation system ensure that all radioactive materials released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The operation of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.12 FUEL STORAGE

Region 1 of the spent fuel storage racks is designed to assure fuel assemblies of less than or equal to 4.1 w/o U-235 enrichment will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in unborated water. These conditions have been verified by criticality analyses.

Region 2 of the spent fuel storage racks is designed to assure fuel assemblies within the burnup and initial enrichment limits of Figure 3.9.2 will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in unborated water. These conditions have been verified by criticality analyses.

The requirement for 1600 ppm boron concentration is to assure the fuel assemblies will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in the event of a postulated accident.