ARKANSAS POWER AND LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE - UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 2
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 January 17, 1975, and March 28, 1975, 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; and

   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.

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

OFFICE ▶						
SURNAME ▶						
DATE ▶					8004220 918	

"(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 2."

3. This license amendment is effective thirty days after the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Zie...nn, Chief
Operating Reactors Branch #2
Division of Reactor Licensing

Attachment:
Change No. 2 to the
Technical Specifications

Date of Issuance: MAY 09 1975

OFFICE ►						
SURNAME ►						
DATE ►						

ATTACHMENT TO LICENSE AMENDMENT NO. 2

CHANGE NO. 2 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Delete pages 13, 14, 15, 16, 19, 20, 23, 24, 37, 38, 39, 48, 48e, 48f, 60, 66, 67a, 68, 71, 72, 73, 73a, 74, 75, 76, 83, 84, 100a from the Appendix A Technical Specifications and insert the attached replacement pages 13, 14, 15, 16, 19, 20, 23, 24, 37, 38, 39, 42a, 48, 48e, 48f, 60, 66, 67a, 68, 71, 72, 73, 73a, 74, 75, 76, 83, 84, 100a. The changed areas on the revised pages are shown by a marginal line.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

The low pressure (1800 psig) and variable low pressure ($16.25T_{out}$ -7756) trip setpoint shown in Figure 2.3-1 have been established to maintain the DNB ratio greater than or equal to 1.3 for those design accidents that result in a pressure reduction. (2,3)

Due to the calibration and instrumentation errors the safety analysis used a variable low reactor coolant system pressure trip value of ($16.25T_{out}$-7796).

D. Coolant outlet temperature

The high reactor coolant outlet temperature trip setting limit (619 F) shown in Figure 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip set point of 620 F.

E. Reactor building pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. Shutdown bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip set point of $\leq$5.0 percent of rated power is automatically imposed during reactor shutdown. 2

2. A high reactor coolant system pressure trip set point of 1720 psig is automatically imposed.

The purpose of the 1720 psig high pressure trip set point is to prevent normal operation with part of the reactor protection system bypassed. This high pressure trip set point is lower than the normal low pressure trip set point so that the reactor must be tripped before the bypass is initiated. The overpower trip set point of $\leq$5.0 percent prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

REFERENCES

(1) FSAR, Section 14.1.2.3

(2) FSAR, Section 14.1.2.2

(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6

Table 2.3-1

Reactor Protection System Trip Setting Limits

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power - 75%)	One Reactor Coolant Pump Operating in Each Loop (Nominal Operating Power - 49%)	Shutdown Bypass
1. Nuclear power, % of rated, max	105.5	105.5	105.5	5.0(3)
2. Nuclear power based on flow(2) and imbalance, % of rated, max	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	1.07 times flow minus reduction due to imbalance(s)	Bypassed
3. Nuclear power based on pump monitors, % of rated, max (4)	NA	NA	55%	Bypassed
4. High reactor coolant system pressure, psig, max	2355	2355	2355	1720(3)
5. Low reactor coolant system pressure, psig, min	1800	1800	1800	Bypassed
6. Variable low reactor coolant system pressure, psig, min	$(16.25T_{out}-7756)$ (1)	$(16.25T_{out}-7756)$ (1)	$(16.25T_{out}-7756)$ (1)	Bypassed
7. Reactor coolant temp, F, max	619	619	619	619
8. gh reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psi

(1) T_{out} is in degrees Fahrenheit (F).
(2) Reactor coolant system flow, %.

(3) Automatically set when other segments of the RPS (as specified) are bypassed.
(4) The pump monitors also produce a trip on: (a) loss of two reactor coolant pumps in one reactor coolant loop, and (b) loss of one or two reactor coolant pumps during two-pump operation. 2

3. LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

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

3.1.1 Operational Components

Specification

3.1.1.1 Reactor Coolant Pumps

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

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant.

3.1.1.2 Steam Generator

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

3.1.1.3 Pressurizer Safety Valves

A. The reactor shall not remain critical unless both pressurizer code safety valves are operable.

B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

Bases

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one half hour or less.(1)

loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rate of 100 F per hour satisfies stress limits for cyclic operation.(2) The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100 F satisfies stress levels for temperatures below the DTT.(3) The plate material and welds in the core region of the reactor vessel have been tested to verify conformity to specified requirements and a maximum NDTT value of 10 F has been determined based on Charpy V-notch tests. The maximum NDTT value obtained for the steam generator shell material and welds was 40 F.

Figures 3.1.2-1 and 3.1.2-2 contain the limiting reactor coolant system pressure-temperature relationship for operation at DTT(4) and below to assure that stress levels are low enough to preclude brittle fracture. These stress levels and their bases are defined in Section 4.3.3 of the FSAR.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT with accumulated nuclear operation. The predicted maximum NDTT increase for the 40-year exposure is shown on Figure 4-10.(4) The actual shift in NDTT will be determined periodically during plant operation by testing of irradiated vessel material samples located in this reactor vessel.(5) The results of the irradiated sample testing will be evaluated and compared to the design curve (Figure 4-11 of FSAR) being used to predict the increase in transition temperature.

The design value for fast neutron (E > 1 Mev) exposure of the reactor vessel is 3.0×10^{10} n/cm^{2}sec at 2568 MWt rated power and an integrated exposure of 3.0×10^{19} n/cm^2 for 40 years operation.(6) The calculated maximum values are 2.2×10^{10} n/cm^{2}sec and 2.2×10^{19} n/cm^2 integrated exposure for 40 years operation at 80 percent load.(4) Figure 3.1.2-1 is based on the design value which is considerably higher than the calculated value. The DTT value for Figure 3.1.2-1 is based on the projected NDTT at the end of the first two years of operation. During these two years, the energy output has been conservatively estimated to be 1.7×10^6 thermal megawatt days which is equivalent to 655 days at 2568 MWt core power. The projected fast neutron exposure of the reactor vessel for the two years is 1.7×10^{18} n/cm^2 which is based on the 1.7×10^6 thermal megawatt days and the design value for fast neutron exposure.

The actual shift in NDTT will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of the vessel in the core area. To compensate for the increases in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the established stress limits during heatup and cooldown.

The NDTT shift and the magnitude of the thermal and pressure stresses are sensitive to integrated reactor power and not to instantaneous power level. Figures 3.1.2-1 and 3.1.2-2 are applicable to reactor core thermal ratings up to 2568 MWt.

The pressure limit line on Figure 3.1.2-1 has been selected such that the reactor vessel stress resulting from internal pressure will not exceed 15 percent yield strength considering the following:

A. A 25 psi error in measured pressure.

B. System pressure is measured in either loop.

C. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations.

For adequate conservatism, in lieu of portions of the Fracture Thoughness Testing Requirements of the proposed Appendix G to 10 CFR 50, a maximum pressure of 550 psig and a maximum heatup rate of 50 F/hr has been imposed below 275 F as shown on Figure 3.1.2-1.

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

The heatup and cooldown rates stated in this specification are intended
as the maximum changes in temperature in one direction in a one hour
period. The actual temperature linear ramp rate may exceed the stated limits
for a time period provided that the maximum total temperature difference 2
does not exceed the limit and that a temperature hold is observed to prevent
the total temperature difference from exceeding the limit for the one hour
period.

REFERENCES

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.10.5

(4) FSAR, Section 4.3.3

(5) FSAR, Section 4.4.5

(6) FSAR, Sections 4.1.2.8 and 4.3.3

3.1.4 Reactor Coolant System Activity

Specification

3.1.4.1 Whenever the reactor is operating under steady-state conditions, the following conditions shall be met.

a. The total specific activity of the primary coolant shall not exceed $72/\overline{E}$ µCi/gm where $\overline{E}$ is the sum of the average beta energy and average gamma energy per disintegration in MEV/disintegration.

b. The I-131 dose equivalent of the radioiodine activity in the primary coolant shall not exceed 3.5 µCi/gm.

c. If the radioactivity in the primary coolant exceeds the limits given above, corrective action shall be taken immediately to return the coolant activity to within these specifications. If the specific activity limits given above cannot be achieved within 24 hours, the reactor shall be brought to a hot shutdown condition using normal operating procedures. If the coolant radioactivity is not reduced to acceptable limits within an additional 48 hours, the reactor shall be brought to a cold shutdown condition and the cause of the out-of-specification operation ascertained.

Bases

Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. 2
Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system.

The parameters assumed in the dose analysis for the single steam generator tube failure included the following values:

1) total primary coolant volume (mass) = 5.2×10^5 lbs.

2) total secondary coolant volume (mass) = 2×10^6 lbs.

3) leakage rate from primary to secondary system = 1 gpm.

4) fission product decay heat energy for 1 hour = 1.56×10^8 BTU.

5) steam mass released to environs = 2.84×10^5 lbs.

6) primary coolant released to secondary (34 minutes) = 8.7×10^4 lbs.

7) minimum primary to secondary iodine equilibrium activity ratio = 20 to 1 (for 1 gpm leakage).

8) specific I-131 dose equivalent activity = 3.5 μCi/gm (Primary)
= 0.17 μCi/gm (Secondary).

9) gross specific activity in primary = $52/\overline{E}$ μCi/gm.

10) $X/Q = 7.0 \times 10^{-4}$ sec/m^3 at limiting point beyond site boundary of 1046 meters for 30 m release height - equivalent to ground level release due to topography including building wake effect for 5 percentile meteorology.

11) total gross radioactivity in primary coolant released to secondary coolant released to environs.

12) ten percent of the combined radioiodine activity from primary activity in secondary coolant and secondary activity present in steam mass (released to environs) assumed released to environs.

The whole body dose resulting from immersion in the cloud containing the released activity would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employed the simple model of the semi-infinite cloud, which gives 2
an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose, because of the short range of beta radiation in air. The resulting whole body dose was determined to be less than 0.5 Rem for this accident.

The thyroid dose from the steam generator tube rupture accident has been analyzed assuming a tube rupture at full load and loss of offsite power at the time of the reactor trip, which results in steam release through the relief valves in the period before the faulty steam generator is isolated and primary system pressure is reduced. The limiting iodine activities for the primary and secondary systems are used in the initial conditions. One-tenth of the iodine contained in the liquid which is converted to steam and passed through the relief valves is assumed to reach the site boundary. The resulting thyroid dose from the combined primary and secondary iodine activity released to the environs was determined to be 1.5 Rem for this accident.

The limit for secondary iodine activity is consistent with the limits on primary system iodine activity and primary-to-secondary leakage of 1gpm. If the activity should exceed the specified limits following a power transient, the major concern would be whether additional fuel defects had developed bringing the total to above expected levels. From the observed removal of excess activity by decay and cleanup, it should be apparent whether activity is returning to a level below the specification limit. Appropriate action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the makeup tank gases to the waste gas decay tanks.

(J) The engineered safety features valves associated with each of the above systems shall be operable or locked in the ES position.

3.3.2 In addition to 3.3.1 above, the following ECCS equipment shall be operable when the reactor coolant system is above 350°F and irradiated fuel is in the core:

(A) Two out of three high pressure injection (makeup) pumps shall be maintained operable, powered from independent essential busses, to provide redundant and independent flow paths.

(B) Engineered safety features valves associated with 3.3.2.a above shall be operable or locked in the ES position.

3.3.3 In addition to 3.3.1 and 3.3.2 above, the following ECCS equipment shall be operable when the reactor coolant system is above 800 psig.

(A) The two core flooding tanks shall each contain an indicated minimum of 13 $\pm$ 0.4 feet (1040 $\pm$ 30 ft^3) of borated water at 600 $\pm$ 25 psig.

(B) Core flooding tank boron concentration shall not be less than 2270 ppm boron.

(C) The electrically operated discharge valves from the core flood tanks shall be open and breakers locked open and tagged.

(D) One of the two pressure instrument channels and one of the two level instrument channels per core flood tank shall be operable.

3.3.4 The reactor shall not be made cricical unless the following equipment in addition to 3.3.1, 3.3.2, and 3.3.3 above is operable.

(A) Two reactor building spray pumps and their associated spray nozzle headers and four reactor building emergency cooling fans and associated cooling units.

(B) The sodium thiosulfate tank shall contain an indicated 31 ft of 30 wt% solution sodium thiosulfate (37,500 lb). The sodium hydroxide tank shall contain an indicated 31 ft. of 20 wt% solution sodium hydroxide (20,500 lb.).

(C) All manual valves in the main discharge lines of the sodium thiosulfate and sodium hydroxide tanks shall be locked open.

(D) Engineered safety feature valves and interlocks associated with 3.3.1, 3.3.2, and 3.3.3 shall be operable or locked in the ES position.

3.3.5 Maintenance shall be allowed during power operation on any component(s) in the high pressure injection, low pressure injection, service water, reactor building spray, reactor building cooling and penetration room 2 ventilation systems which will not remove more than one train of each system from service. Maintenance shall not be performed on components which would make the affected system train inoperable for more than 24 consective hours. Prior to initiating maintenance on any component of a train in any system, the redundant component of that system shall be demonstrated to be operable within 24 hours prior to the maintenance.

3.3.6 If the conditions of Specifications 3.3.1, 3.3.2, 3.3.3, 3.3.4 and 3.3.5 cannot be met except as noted in 3.3.7 below, reactor shutdown shall be initiated and the reactor shall be in hot shutdown condition within 36 hours and, if not corrected, in cold shutdown condition within an additional 72 hours.

3.3.7 Exceptions to 3.3.6 shall be as follows: 2

(A) If the conditions of Specification 3.3.1(G) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other BWST level instrument channel shall be operable.

(B) If the conditions of Specification 3.3.3(D) cannot be met, reactor operation is permissible only during the succeeding seven days unless such components are sooner made operable, provided that during such seven days the other CFT instrument channel (pressure or level) shall be operable.

Bases

The requirements of Specification 3.3.1 assure that below 350°F, adequate long term core cooling is provided. Two low pressure injection pumps are specified. However, only one is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident.

The post-accident reactor building cooling and long-term pressure reduction may be accomplished by four cooling units, by two spray units or by a combination of two cooling units and one spray unit. Post-accident iodine removal may be accomplished by one of the two spray system strings. The specified requirements assure that the required post-accident components are available for both reactor building cooling and iodine removal. Specification 3.3.1 assures that the required equipment is operational.

The borated water storage tank is used for three purposes:

(A) As a supply of borated water for accident conditions.

(B) As an alternate supply of borated water for reaching cold shutdown.(2)

(C) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(3)

350,000 gallons of borated water are supplied for emergency core cooling and reactor building spray in the event of a loss-of-coolant accident. This amount fulfills requirements for emergency core cooling. 16,000 gallons of borated water are required to reach cold shutdown. The borated water storage tank capacity of 380,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent crystallization and local freezing of the boric acid. The boron concentration is set at a value that will maintain the core at least 1 percent $\Delta k/k$ subcritical at 70°F without any control rods in the core. The concentration for 1% $\Delta k/k$ subcriticality is 1609 ppm boron in the core, while the minimum value specified in the borated water storage tank is 2270 ppm boron.

Specification 3.3.2 assures that above 350°F two high pressure injection pumps are also available to provide injection water as the energy of the reactor coolant system is increased.

Specification 3.3.3 assures that above 800 psig both core flooding tanks are operational. Since their design pressure is 600 ± 25 psig, they are not brought into the operational state until 800 psig to prevent spurious injection of borated water. Both core flooding tanks are specified as a single core flood tank has insufficient inventory to reflood the core.[1]

Specification 3.3.4 assures that prior to going critical the redundant reactor building cooling unit and spray are operational.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

When the reactor is critical, maintenance is allowed per Specification 3.3.5. Operability of the specified components shall be based on the results of testing as required by Technical Specification 4.5. The maintenance period of up to 24 hours is acceptable if the operability of equipment redundant to that removed from service is demonstrated within 24 hours prior to removal. 2
Exceptions to Specification 3.3.6 permit continued operation for seven days if one of two BWST level instrument channels is operable or if either the pressure or level instrument channel in the CFT instrument channel is operable.

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severence, limit the peak clad temperature to less than 2300°F and the metal-water reaction to that representing less than 1 percent of the clad.

The service water system consists of two independent but interconnected, full capacity, 100% redundant systems, to ensure continuous heat removal.[4]

One service water pump is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

3.5.1.7 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. The relief valve setting for the DHR system shall be equal to or less than 450 psig. 2

3. Except for physics tests or exercising control rods, the control rod withdrawal limits are specified on Figures 3.5.2-1A and 3.5.2-1B for four pump operation and on Figure 3.5.2-2 for three or two pump operation. If the control rod position limits are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within four hours.

4. Except for physics tests, power shall not be increased above the power level cutoff (see Figures 3.5.2-1) unless the xenon reactivity is within 10 percent of the equilibrium value for operation at rated power and asymptotically approaching stability.

3.5.2.6 Reactor Power Imbalance shall be monitored on a frequency not to exceed two hours during power operation above 40 percent rated power. Except for physics tests, imbalance shall be maintained within the envelope defined by Figure 3.5.2-3. If the imbalance is not within the envelope defined by Figure 3.5.2-3, corrective measures shall be taken to achieve an acceptable imbalance. If an acceptable imbalance is not achieved within four hours, reactor power shall be reduced until imbalance limits are met.

3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the superintendent.

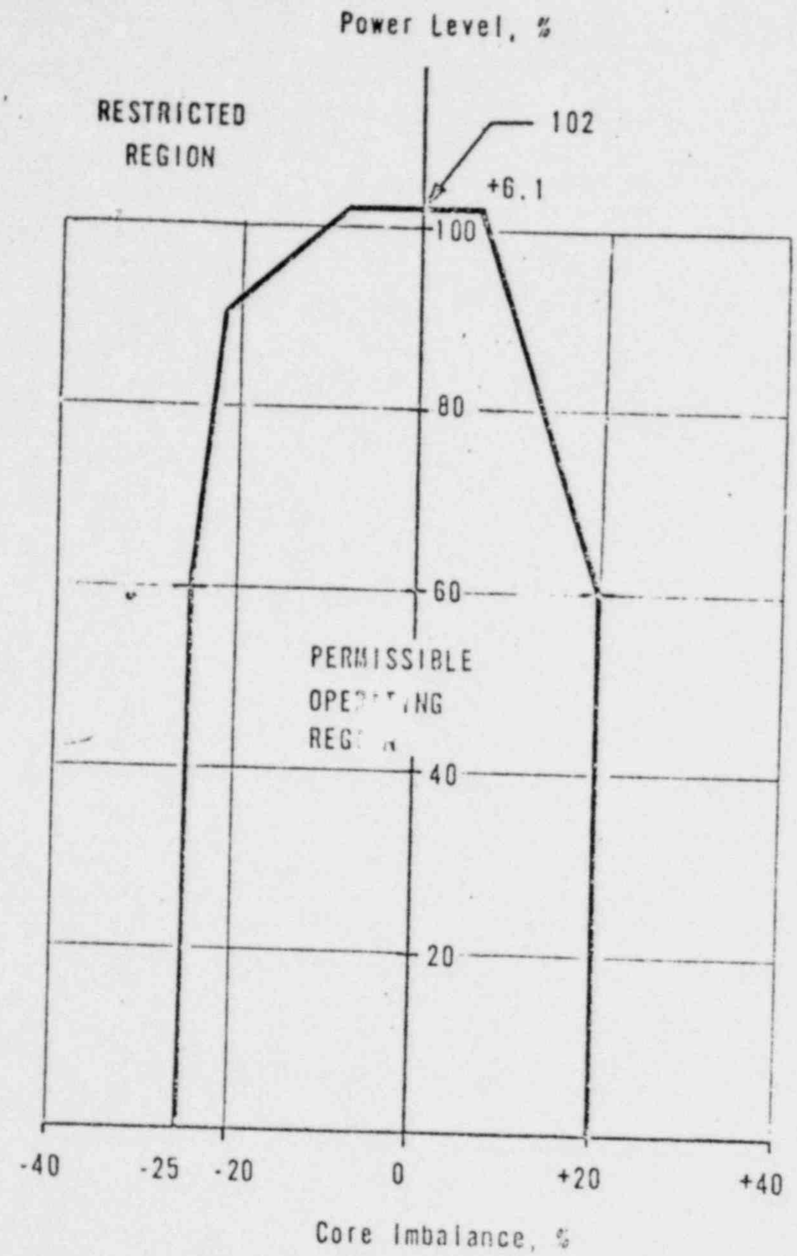
Bases

The power-imbalance envelope defined in Figure 3.5.2-3 is based on 1) LOCA
analyses which have defined the maximum linear heat rate (see Figure 3.5.2-4)
such that the maximum clad temperature will not exceed the Interim Acceptance
Criteria and 2) the Protective System Maximum Allowable Setpoints (Figure 2
2.3-2). Corrective measures will be taken immediately should the indicated
quadrant tilt, rod position, or imbalance be outside their specified boundary. Operation in a situation that would cause the interim acceptance criteria to be approached should a LOCA occur is highly improbable because all of the power distribution parameters (quadrant tilt, rod position, and imbalance) must be at their limits while simultaneously all other engineering and uncertainty factors are also at their limits.* Conservatism is introduced by application of:

a. Nuclear uncertainty factors
b. Thermal calibration
c. Fuel densification effects
d. Hot rod manufacturing tolerance factors

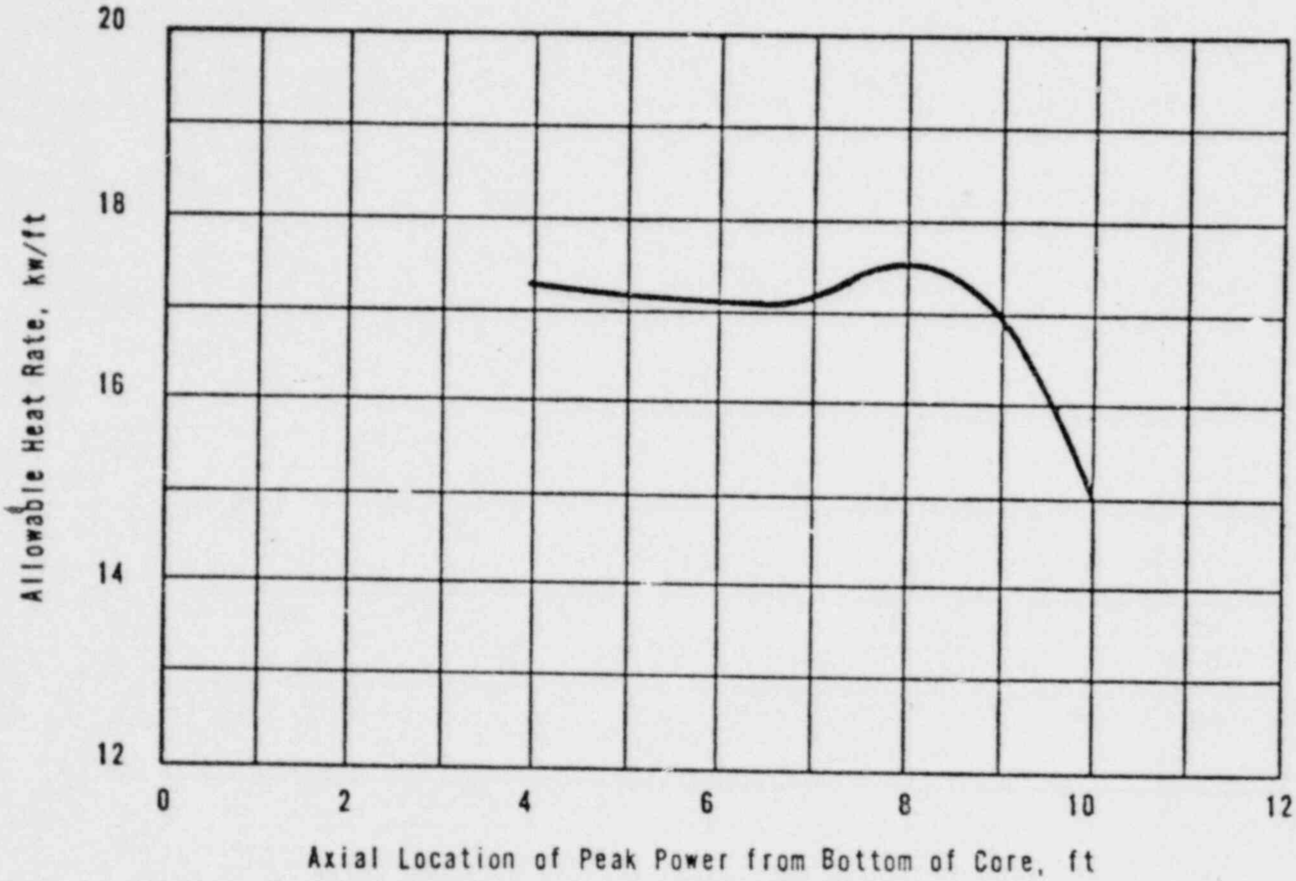
The 30 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower part of the stroke. Control rods are arranged in groups or banks defined as follows:

*Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument and calibration errors. The method used to define the operating limits is defined in plant operating procedures.



ARKANSAS POWER & LIGHT CO ARKANSAS NUCLEAR ONE-UNIT 1	OPERATIONAL POWER IMBALANCE ENVELOPE	FIG. NO. 3.5.2-3

2



ARKANSAS POWER & LIGHT CO. ARKANSAS NUCLEAR ONE-UNIT 1	LOCA LIMITED MAXIMUM ALLOWABLE LINEAR HEAT RATE	FIG. NO. 3.5.2-4

3.9 RADIOACTIVE DISCHARGE

This specification has been replaced by specification 2.4 of the environmental technical specifications (Appendix B to the operating license).

 (Next page is 66)

3.10 SECONDARY SYSTEM ACTIVITY

Applicability

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

Objective

To limit the maximum secondary system activity.

Specification

The I-131 dose equivalent of the radioiodine activity in the secondary coolant shall not exceed 0.17 μCi/gm.

Bases

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

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

The dose analysis performed to determine the maximum allowable reactor coolant activity assuming the maximum allowable primary to secondary leakage of 1 gpm as given in the Bases for Specification 3.1.4.1 indicated that the controlling accident to determine the allowable secondary coolant activity would be the rupture of a steam generator tube. For the loss of load incident with a loss of 205,000 pounds of water released to the atmosphere via the relief valves, the resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be 0.6 Rem with the same meteorological and iodine release assumptions used for the steam generator tube rupture as given in the Bases for Specification 3.1.4.1. For the less probable accident of a steam line break, the assumption is made that a loss of 1×10^6 pounds of water or the contents of one loop in the secondary coolant system occurs and is released directly to the atmosphere. Since the water will flash to steam, the total radioiodine activity is assumed to be released to the atmosphere. The resulting thyroid dose at the I-131 dose equivalent activity limit of 0.17 μCi/gm would be less than 28 Rem with the same meteorological assumptions used for the steam generator tube rupture and loss of load incident. 2

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During non-steady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once each refueling period for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

The rotation schedule for the reactor protective channels is as follows:

Channels A, B, C, D	Before Startup if shutdown greater than 24 hours
Channel A	One Week After Startup
Channel B	Two Weeks After Startup
Channel C	Three Weeks After Startup
Channel D	Four Weeks After Startup

The reactor protective system instrumentation test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every four weeks. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation. | 2

REFERENCE

FSAR Section 7.1.2.3.4

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
20. Reactor Building Spray System Logic Channels	NA	M (1)	NA	(1) Including RB spray pump, spray valve, and chem. add. valve logic channels.
21. Reactor Building Spray System Analog Channels				
a. Reactor Building Pressure Channels	NA	M	R	
22. Pressurizer Temperature Channels	S	NA	R	
23. Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.
24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator.
25. Core Flooding Tanks				
a. Pressure Channels	S	NA	R	
b. Level Channels	S	NA	R	
26. Pressurizer Level Channels	S	NA	R	
27. Makeup Tank Level Channels	D	NA	R	
28. Radiation Monitoring Systems	W	M(1)	Q(2)	(1) Check functioning of self-checking feature on each detector. (2) R for those detectors inaccessible during normal operation
29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	

Table 4.1-1 (Cont'd)

Channel Description	Check	Test	Calibrate	Remarks
30. Decay Heat Removal System Isolation Valve Automatic Closure And Interlock System	S(1)(2)	M(1)(3)	R	(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) Shall Also Be Tested During Refueling Shutdown Prior to Re-pressurization at a pressure greater than 300 but less than 420 psig. 2
31. Turbine Overspeed Trip Mechanism	N/A	R	N/A	
32. Steam Line Break Instrumentation And Control		(Later)		
33. Diesel Generator Protective Relaying, Starting Interlocks And Circuitry	M	Q	N/A	
34. Off-site Power Under-voltage And Protective Relaying Interlocks And Circuitry	W	R	R	
35. Borated Water Storage Tank Level Indicator	W	N/A	R	
36. Boric Acid Mix Tank				
a. Level Channel	N/A	N/A	R	
b. Temperature Channel	M	N/A	R	

Table 4.1-2

Minimum Equipment Test Frequency

Item	Test	Frequency
1. Control Rods	Rod Drop Times of All Full Length Rods 1/	Each Refueling Shutdown
2. Control Rod Movement	Movement of Each Rod	Every Two Weeks Above Cold Shutdown Conditions
3. Pressurizer Code Safety Valves	Setpoint	One Within 2 Weeks Prior to or Following Each Refueling Shutdown
4. Main Steam Safety Valves	Setpoint	Four Within 2 Weeks Prior to or Following Each Refueling Shutdown
5. Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown
6. Reactor Coolant System Leakage	Evaluate	Daily
7. Charcoal and High Efficiency Filters in Control Room, Penetration Room Ventilation System, Hydrogen Purge System, and Reactor Purge System	Charcoal and HEPA Filter for Iodine and Particulate Removal Efficiencies. DOP Test on HEPA Filters. Freon Test on Charcoal Filter Units 2/	Each Refueling Period and at Any Time Work on Filters Could Alter Their Integrity
8. Reactor Building Isolation Trip	Functioning	Each Refueling Shutdown
9. Service Water Systems	Functioning	Each Refueling Shutdown
10. Spent Fuel Cooling System	Functioning	Each Refueling Shutdown Prior to Use
11. Decay Heat Removal System Isolation Valve Automatic Closure and Isolation System	Functioning	Each Refueling Shutdown Prior to Repressurization at a pressure greater than 300 psig but less than 420 psig. 2

1/ Same as tests listed in section 4.7
2/ Same as tests listed in sections 4.4.3, 4.5.3, 4.11 and 4.12

Table 4.1-2 (Continued)
Minimum Equipment Test Frequency

	Item	Test	Frequency
12.	Flow Limiting Annulus on Main Feedwater Lines at Reactor Building Penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test.

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency	
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	
	b. Gross Activity Determination	b. 3 times/week and at least every third day(1)(6)(7)	
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	2
	d. Dissolved Gases	d. Weekly (7)	
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	
	f. Boron Concentration	f. 3 times/week	
	g. Radiochemical Analysis for $\bar{E}$ Determination (2)(4)	g. Monthly (7)	
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup	
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup	
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)	
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup	2
7. Sodium Thiosulfate Tank Sample	Sodium Thiosulfate Concentration	Quarterly and after each makeup	

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 μCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of $\overline{E}$. A radiochemical analysis and calculation of $\overline{E}$ and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes.

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level.

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity.

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2.

(6) Whenever the steady state radioiodine or gross radioactivity concentration 2
of prior operation is greater than 1 percent but less than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above.

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition.

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition.

(9) Required only when fuel is in the pool and prior to transferring fuel to the pool.

(10) Not required when not generating steam in the steam generators.

4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the reactor coolant system pressure boundary.

Objective

To assure the continued integrity of the reactor coolant system pressure boundary.

Specification

4.2.1 Prior to initial unit operation, an ultrasonic test survey shall be made of reactor coolant system pressure boundary welds as required to establish preoperational integrity and base line data for future inspections.

4.2.2 Post operational inspections of components shall be made in accordance with the methods and intervals indicated in IS-242 and IS-261 of Section XI of the ASME Boiler and Pressure Vessel Code, 1971, including 1971 Winter addenda, except as follows:

IS-261 Item	Component	Exception
1.4	Primary Nozzle to Vessel Welds	1 RC inlet nozzle to be inspected after approx. 3 1/3 years operation. All four RC inlet nozzles to be inspected at or near the end of interval. Both RC outlet nozzles will be inspected after approx. 6 2/3 yrs. operation. One core flood nozzle will be inspected after 3 1/3 years operation and one core flood nozzle inspected near the end of interval
3.3	Safe Ends on Heat Exchanger	Not Applicable
4.1	Vessel Safe End Welds	Not Applicable
4.2	Valve Pressure Retaining Bolting Larger than 2"	Not Applicable
4.9	Integrally Welded Supports	Not Applicable
6.1	Valve Body Welds	Not Applicable
6.3	Valve to Safe End Welds	Not Applicable

4.4.1.2.5 Test Frequency

Local leak detection tests shall be performed at a frequency of at least each refueling period, but in no case at intervals greater than two years except that:

(a) The equipment hatch and fuel transfer tube seals shall be additionally tested after each opening.

(b) If a personnel hatch or emergency hatch door is opened when reactor building integrity is required, the affected door seal shall be tested. In addition, a pressure test shall be performed on the personnel and emergency hatches every six months. 2

4.4.1.3 Reactor Building Modifications

Any major modification or replacement of components affecting the reactor building integrity shall be followed by either an integrated leak rate test or a local leak test, as appropriate, and shall meet the acceptance criteria specified in 4.3.1.1 and 4.3.1.2 respectively.

4.4.1.4 Isolation Valve Functional Tests

Every three months, remotely operated reactor building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during plant operation. The latter valves shall be tested during each refueling period.

4.4.1.5 Visual Inspection

A visual examination of the accessible interior and exterior surfaces of the reactor building structure and its components shall be performed during each refueling shutdown and prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the reactor building's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, nondestructive tests, and inspections, and local testing where practical prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

Bases (1)

The reactor building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 285 F. Prior to initial operation, the reactor building will be strength tested at 115% of design pressure and leak rate tested at the design pressure. The reactor building will also be leak tested prior to initial operation at not less than 50% of the design pressure. These tests will verify that the leakage rate from reactor building pressurization satifies the relationships given in the specification.

The performance of a periodic integrated leakage rate test during plant life provides a current assessment of potential leakage from the reactor building in case of an accident that would pressurize the interior of the reactor building. In order to provide a realistic appraisal of the integrity of the reactor building under accident conditions, the reactor building isolation valves are to be closed in the normal manner. The test pressure of 30 psig for the periodic integrated leakage rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the pre-operational leakage rate test at 30 psig. The specification provides a relationship for relating the measured leakage of air at 30 psig to the potential leakage at 59 psig. The frequency of the periodic integrated leakage rate test is keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete reactor building to a 0.20% leakage rate at 59 psig during pre-operational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at design pressure, of those portions of the reactor building envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value of $.60L_a$ leakage that is specified as acceptable from tested penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that an important part of the structural integrity of the reactor building is maintained.

<u>References</u>

(1) FSAR, Sections 5 and 13.

4.6.2 Station Batteries and Switchyard Batteries

1. The voltage, temperature and specific gravity of a pilot cell in each bank and the overall battery voltage of each bank shall be measured and recorded daily.

2. Measurements shall be made quarterly of voltage of each cell to the nearest 0.01 volt of the specific gravity of each cell, and of the temperature of every fifth cell in each bank. The level of the electrolyte shall be checked and adjusted as required. All data, including the amount of water added to any cell, shall be recorded.

3. During each refueling outage, a performance discharge test shall be conducted in accordance with the manufacturer's instructions, for the purpose of determining battery capacity.

4. Any battery charger which has not been loaded while connected to its 125V d-c distribution system for at least 30 minutes during every quarter shall be tested and loaded while connected to its bus for 30 minutes. The third battery charger, which is capable of being connected to either of the two 125V d-c distribution systems, shall be loaded while connected to each bus for at least 30 minutes every quarter. 2

4.6.3 Emergency Lighting

The correct functioning of the emergency lighting system shall be verified at least once each year.

resulting from the other two postulated accidents were determined. Since we consider the probability of occurrence for either the steam generator tube rupture or the loss of load incident to be comparable, the acceptable thyroid dose limit for either incident was taken as 1.5 Rem. We consider the occurrence of a steam line break accident outside containment to be more likely than a loss of coolant accident but less probable than either the steam generator tube rupture or the loss of load incident. For this reason, we consider the acceptable thyroid dose limit to be 30 Rem or significantly less than the guideline doses of 10 CFR Part 100.

As stated in the bases for this section of the technical specifications, the resulting thyroid doses using the specified secondary system activity limit of 0.17 µCi/gm of I-131 dose equivalent are approximately 1.5 Rem for the steam generator tube rupture (as stated in the Bases to Specification 3.1.4), 0.6 Rem for the loss of load incident and 28 Rem for the steam line break accident outside containment. All of these doses are less than the above stated dose guidelines for these accidents and indicate that the controlling accident for determining the secondary coolant radioiodine limit is the steam generator tube rupture. An increase in the ratio of radioiodine specific activity for the reactor coolant to the secondary coolant would directly reduce the calculated dose for the two accidents involving only secondary coolant releases. However, an increase in this ratio would not significantly reduce the calculated dose for the steam generator tube rupture which releases both reactor coolant and secondary coolant radioactivity.

(7) Table 4.1-1 (Item 30) and Table 4.1-2 (Item 11) - Note 3 has been changed to reflect the newly established setpoints on the isolation valves of DHRS given in Specification 3.5.1.7 and gives the pressure range within which the test must be performed. The test will verify the correct setpoints for the isolation valves. The same change was made to the test frequency column in Table 4.1-2 for consistency.

(8) Table 4.1-3 - The minimum sampling and analysis frequency and tests have been changed for the reactor coolant samples. The Gross Activity Determination (previously designated as Gross Beta and Gamma Activity) frequency has been reduced from 5 times per week to 3 times per week and at least every third day. This frequency also has been designated for measuring the Chemistry and Boron Concentration in the Reactor Coolant. Experience has shown that such frequencies are adequate to detect changes in coolant chemistry on a timely basis and permits reactor operation over a week-end or holiday without the need for a reactor coolant sample analysis. Gamma Isotopic Analysis frequency has been increased from monthly

to bi-weekly (once every two weeks) and the Radiochemical Analysis for E Determination has been increased from semi-annually to monthly. Both of these changes in frequency are to detect on a timely basis any change in quality of the gross radioactivity contained in the reactor coolant. Experience has shown that such frequencies will detect changes in quality of radioactivity due to additional failed fuel or change in reactor operations. The Gross Radioiodine Determination has been added to detect radioiodine activity levels in the reactor coolant for compliance with Section 3.1.4 requirements. The specified frequency for the analysis is weekly but shall be more frequent if the gross activity increases by a given amount as specified by Note 3. Experience has shown that a weekly frequency with this condition for more frequent analysis is adequate to detect on a timely basis any changes in reactor coolant radioiodine levels.

A determination of dissolved gases concentration in the reactor coolant is required by Specification 3.1.9.1 which places a limit of 100 std cc per liter of water of dissolved gases in the reactor coolant for control rod operation. The buildup of dissolved gases in the reactor coolant is a slow process and therefore weekly determination of this parameter is considered adequate for timely detection of any unusual increases.

The existing requirements for determining tritium concentration, Sr-89 and Sr-90 concentration and gross alpha activity in the reactor coolant are not required because there are no limits necessary for these radioisotopes for operating of the facility. The existing requirement for determining gross beta-gamma activity in the secondary coolant is not required because such activity would not be present and no limits on operation for gross activity are required. Therefore, these requirements have been deleted from the table. Analyses for radioactivity levels and dissolved gases are not required when the plant is in the cold or refueling shutdown condition because these parameters do not affect the safety of the plant when in these shutdown conditions. Thus, Note 7 provides for these exclusions. To determine the level and duration of possible radioactivity spiking for both gross and iodine activity, additional analysis depending upon level of radioactivity during the previous steady state operation at the time of reactor startup and during reactor startup is required by Note 6. This note applies only to the gross determinations of total activity and radioiodines. Note 1 basically remains the same as previously stated for this table except that the increased frequency of analysis is required only until steady state activity level is established. As discussed in Note 1 and Note 2, the gross activity determination

OFFICE ➤						
SURNAME ➤						
DATE ➤						

will be based on the activity present 15 minutes after sampling. This time period is equivalent to the minimum expected decay time from the release point to the nearest site boundary in case of a steam generator tube rupture. Note 2, which is associated with the determination of Ē, specifies the method to be used for the determination of Ē, the frequency of determination for Ē and radioiodine, and the reference (or equivalent) source to be used to determine the individual gamma and beta energies per disintegration for the radioisotopes present in the reactor coolant. Note 4 to this same sample determination indicates that all radioiodine activities (I-131, 132, 133, 134, and 135) are to be weighted to determine the I-131 dose equivalent activity actually present in the reactor coolant for comparison with the limit established in Specification 3.1.4.1.b. Note 8 states that the O_2 analysis is not required when the plant is in the cold or refueling shutdown condition for the same reason as given for Note 7 acceptability.

Note 9 states that a determination of boron concentration in the Spent Fuel Pool is required only if fuel is present in the pool and prior to fuel being transferred to the pool. Since the boron in pool water is to assure that the fuel remains subcritical, it is not required when there is no fuel in the pool; therefore, Note 9 is acceptable. Notes 5 and 10 apply to the secondary coolant sampling and analysis program. Note 5 requires additional sampling and analysis if the primary to secondary leakage increases significantly. Note 10 eliminates the requirement for sampling and analysis of the secondary system coolant when steam generation is not occurring. If steam is not being generated, the postulated accidents, which established the secondary coolant activity limit, could not occur or would not result in any significant release of radioiodine to the environs from the secondary coolant. Note 4 also applies to the secondary coolant radioiodine activity determination for assessing the I-131 dose equivalent activity for comparison with the limit established in Specification 3.10. Note 7 also applies to the secondary coolant sampling and analysis when the plant is in the cold or refueling shutdown condition.

(9) Specification 4.4.1.2.5(b) - To make the testing requirements on the personnel hatch and emergency hatch door seals consistent with Appendix J of 10 CFR Part 50, the first sentence of this specification was modified. The addition of the phrase "when reactor building integrity is required" satisfies Appendix J, 10 CFR 50, requirements provided the existing phrase, 'but no more frequently than daily during normal operation' is deleted. The existing requirement for weekly testing during refueling or cold shutdowns

OFFICE ➤						
SURNAME ➤						
DATE ➤						

is not necessary for the proposed specification which limits testing requirements to times when building integrity is required. These changes are acceptable and were made.

(10) Specification 4.6.2.4 - The existing testing requirement relating to the third battery charger has been expanded to apply to all three battery chargers to assure that adequate surveillance of all battery chargers is provided. The additional surveillance requirement is considered appropriate and acceptable to the staff and should increase the reliability of the station battery system.

(11) Appropriate changes in the Bases were made for clarification, but they do not affect the specifications governing operation of the facility.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the change 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 change 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: MAY 09 1975

OFFICE ➤						
SURNAME ➤						
DATE ➤						