Docket File
DCS MS-016

NOV 10 1983

Docket No. 50-368

Mr. John M. Griffin, Vice President
Nuclear Operations
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Griffin:

The Commission has issued Amendment No. 49 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2), in partial response to your application dated February 23, 1983 as supplemented by letter dated April 18, 1983. The amendment consists of changes to the Technical Specifications (TS) as listed below.

1. Corrections of several typographical errors in Sections 3 and 4 of the TS.

2. Changes to TS Bases 2.2.1 to be consistent with the corresponding specifications which were revised with the issuance of Amendment No. 24.

3. Update of TS Bases 2.2.2 to reflect the final submittal of a reference document pertaining to the Core Protection Calculator (CPC) addressable constants.

4. Change to TS 4.4.9.2 to delete the surveillance requirement pertaining to the pressurizer spray water temperature differential (as the requirement is adequately addressed in Section 5.7 of the TS.

The changes which were proposed to reflect the reorganization of the Energy Supply Department of Arkansas Power and Light Company are being reviewed separately.

We have determined that the proposed change pertaining to the incore detector surveillance requirements is not needed. This has been discussed with and agreed to by your staff.

A copy of the Safety Evaluation in enclosed. The Notice of Issuance will be included in the Commission's next monthly Federal Register notice.

Sincerely,

Robert S. Lee

Robert S. Lee, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 49 to NPF-6
2. Safety Evaluation

cc w/enclosures:
See next page

~~Contents~~ my concurrence is contingent on there being no problems with Part A of the licensee area because only a portion is being acted on now.

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Immediately before issuing amend. ~~[illegible]~~ check to see that there are no comments or requests for hearing. If there are report back to OELD before issuance

Also see editorial change on Fed Reg notice

ORB#3:DL	ORB#3:DL	ORB#3:DL	AD:OR:DL	OELD
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10/13 McGurren See clip for typo.

M Wh

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 49
License No. NPF-6

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

   A. The application for amendment by Arkansas Power & Light Company (the licensee) dated February 23, 1983 as supplemented April 18, 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 (1) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, except where otherwise stated in specific license conditions.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 10, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 49

FACILTIY 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. The corresponding overleaf pages are provided to maintain document completeness.

Pages

3/4 2-7
3/4 2-13
3/4 2-14
3/4 3-5a

3/4 4-25
B 2-3
B 2-4
B 2-7
B 3/4 4-10

BASES

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at $\leq$ 2370.887 psia which is below the nominal lift setting (2500 psia) of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at $\geq$ 1712.757 psia. This trip's setpoint may be manually decreased, to a minimum value of 100 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at $\leq$ 200 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately 900 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at $\leq$ 200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in CEN-44(A)P, "Core Protection Calculator Functional Description," January 7, 1977, Supplement 1P, May 13, 1977, Supplement 2P, May 19, 1977, Supplement 3P, September 2, 1977; CEN-45(A)P, "Control Element Assembly Calculator Functional Description," January 7, 1977; CEN-53(A)P, "ANO-2 Cycle 1 CPC and CEAC Data Base Document," May 20, 1977, Amendment 1P, June 28, 1977, Supplement 2P, September 2, 1977.

Manual Reactor Trip

The Manual Reactor Trip is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA. This trip initiates a reactor trip at a linear power level of $\leq$ 110.712% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of $\leq$ 0.819% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

BASES

a.	RCS Cold Leg Temperature-Low	$\geq$ 465°F
b.	RCS Cold Leg Temperature-High	$\leq$ 605°F
c.	Axial Shape Index-Positive	Not more positive than +0.6
d.	Axial Shape Index-Negative	Not more negative than -0.6
e.	Pressurizer Pressure-Low	$\geq$ 1750 psia
f.	Pressurizer Pressure-High	$\leq$ 2400 psia
g.	Integrated Radial Peaking Factor-Low	$\geq$ 1.28
h.	Integrated Radial Peaking Factor-High	$\leq$ 4.28
i.	Quality Margin-Low	$\geq$ 0

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

2.2.2 CPC Addressable Constants

The Core Protection Calculator (CPC) addressable constants are provided to allow calibration of the CPC system to more accurate indications such as calorimetric measurements for power level and RCS flowrate and incore detector signals for axial flux shape, radial peaking factors and CEA deviation penalties. Other CPC addressable constants allow penalization of the calculated DNBR and LPD values based on measurement uncertainties or inoperable equipment. Administrative controls on changes and periodic checking of addressable constant values (see also Technical Specifications 3.3.1.1 and 6.8.1) ensures that inadvertent misloading is unlikely. The methodology for determination of CPC addressable constant values is described in MSS-NA2-P, "Arkansas Nuclear One-Unit 2 Core Protection Calculator Addressable Constant Determination Methodology" dated August 1981.

DNBR MARGIN

LIMITING CONDITION FOR OPERATION

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-3 or 3.2-4, as applicable.

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

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

b. Be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above 20% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-4.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.4.4 The following DNBR penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days:

Burnup $\left(\frac{GWD}{MTU}\right)$	DNBR Penalty (%)
0-30	2.0
30-40	3.5
40-50	5.5

The penalty for each batch will be determined from the batch's maximum burnup assembly and applied to the batch's maximum radial power peak assembly. A single net penalty for COLSS and CPC will be determined from the penalties associated with each batch, accounting for the offsetting margins due to the lower radial power peaks in the higher burnup batches.

AXIAL SHAPE INDEX

LIMITING CONDITION FOR OPERATION

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

a. COLSS OPERABLE
$-0.28 \leq ASI \leq +0.28$

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

*See Special Test Exception 3.10.2.

PRESSURIZER PRESSURE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 1

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

b. With both CEACs inoperable, operation may continue provided that:

   1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to $\geq$ 11% of RATED THERMAL POWER.

   2. Within 4 hours:

      a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

      b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

      c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

   3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 2. a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in its group.

ACTION 6 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	$\leq$ 0.40 seconds*
3. Logarithmic Power Level - High	$\leq$ 0.40 seconds*
4. Pressurizer Pressure - High	$\leq$ 0.90 seconds
5. Pressurizer Pressure - Low	$\leq$ 0.90 seconds
6. Containment Pressure - High	$\leq$ 1.59 seconds
7. Steam Generator Pressure - Low	$\leq$ 0.90 seconds
8. Steam Generator Level - Low	$\leq$ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ 2.58 seconds*
b. CEA Positions	$\leq$ 1.58 seconds**

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 200°F in any one hour period, and

b. A maximum cooldown of 200°F in any one hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cool-down.

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 AND 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: ALL MODES

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

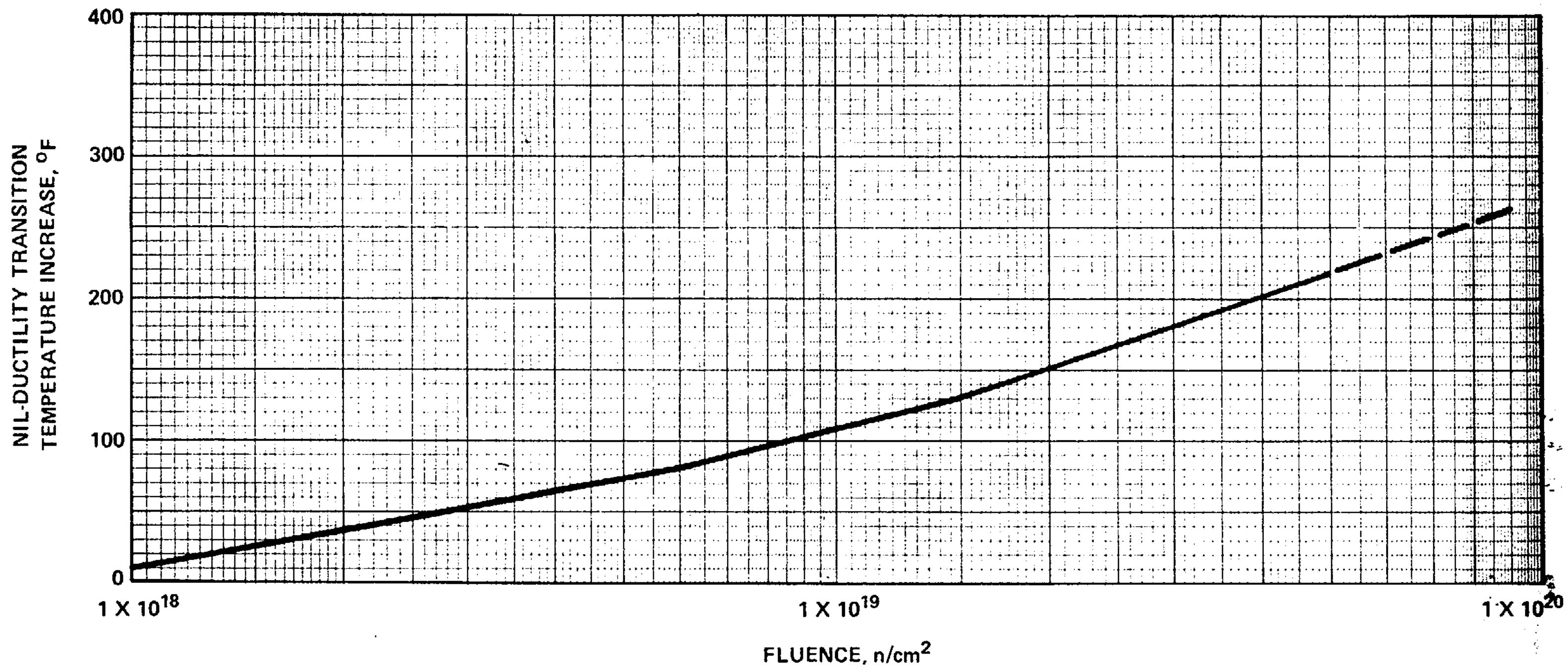

Figure B3/4.4-1
Nil-Ductility Transition Temperature Increase
as a Function of Fast (E > 1mev) Neutron Fluence (550°F Irradiation)

BASES

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 49 TO FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT 2

DOCKET NO. 50-368

Introduction

By letters dated February 23 and April 18, 1983, Arkansas Power & Light Company (the licensee or AP&L) requested an amendment to Facility Operating License No. NPF-6 for operation of Arkansas Nuclear One, Unit 2 (ANO-2), located in Pope County, Arkansas. This Safety Evaluation (SE) addresses a number of changes to the Technical Specifications (TS) as discussed in each of the following sections. In addition, the basis for denying the proposed change to the incore detector surveillance requirements is provided. The changes which were proposed to reflect the reorganization of the Energy Supply Department of Arkansas Power and Light Company are being reviewed separately.

1. Pressurizer Spray Water Temperature Differential

The licensee proposed to revise TS 4.4.9.2 and the corresponding TS bases pertaining to a surveillance of the pressurizer spray water temperature differential. The proposed changes would delete the aforestated surveillance requirement from TS 4.4.9.2 and revise the corresponding TS bases accordingly.

In support of the proposed changes, the licensee states that the pressurizer spray water temperature differential limits are addressed elsewhere in the TS, namely, Section 5.7, "Component Cyclic or Transient Limits". Moreover, the licensee notes the fact that the pressurizer spray water temperature differential, per se, is not a limiting condition for operation.

We have reviewed the proposed changes and find them acceptable based on the fact that Section 5.7 of the TS adequately addresses the pressurizer spray water temperature differential limits to assure that the pressurizer is operated within the design criteria.

2. Miscellaneous Changes

- Amendment No. 24 to the license issued on June 19, 1981 revised the allowable values for the reactor protective instrumentation trip set-point limits for the Linear Power Level-High to 110.712%, the Pressurizer Pressure-High to 2370.887 psia and the Pressurizer Pressure-Low to 1712.757 psia. However, the changes were not reflected in the corresponding TS bases. Therefore, with the issuance of this amendment, TS Bases 2.2.1 is revised to reflect the aforestated changes.

- The licensee submitted an interim document describing the methodology for determination of the Core Protection Calculator (CPC) addressable constant values on May 26, 1981 and committed to provide a final document by August 17, 1981. The interim document is referenced in the TS Bases 2.2.2. Subsequently, in August 1981, the licensee submitted the final document. Therefore, with the issuance of this amendment, the final document would be referenced in TS Bases 2.2.2.

- Four typographical errors which have been identified would be corrected with the issuace of this amendment.

3. Incore Detector Surveillance

The licensee proposed to change TS 4.3.3.2.a which states that the incore detection system shall be demonstrated OPERABLE by performance of a CHANNEL CHECK within 24 hours prior to its use <u>and</u> at least once per 7 days when required for monitoring the azimuthal power tilt, radial peaking factors, local power or DNB margin. The proposed change would replace "and" with "or" in this specification.

We have reviewed the proposed change and found, based on our discussion with the licensee, that the change was requested in order to avoid performing a channel check within 24 hours prior to performing a periodic surveillance if the system was in continuous use and channel checks were being performed every 7 days. It is our interpretation of the present TS (with "and") that, if the system is in continuous operation, a channel check every 7 days is sufficient. There is no need to perform additional checks. We have discussed this interpretation with the licensee and he has agreed that the proposed change is not needed. On this basis, we are not approving the proposed change.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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: NOV 10 1983

Principal Contributors:
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