Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

March 12, 1998

SUBJECT: ISSUANCE OF AMENDMENT NO. 189 TO FACILITY OPERATING LICENSE NO. NPF-6 - ARKANSAS NUCLEAR ONE, UNIT NO. 2 (TAC NO. M99654)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment No. 189 to Facility Operating License No. NPF-6 for the Arkansas Nuclear One, Unit No. 2 (ANO-2). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 23, 1997, as supplemented by the letters dated February 27 and March 4, 1998.

The amendment changes the Reactor Protective System (RPS) and Engineering Safety Actuation System (ESFAS) trip set point and allowable values for steam generator low pressure. The amendment also relocates the RPS and ESFAS response time tables from the Technical Specifications to the Safety Analysis Report as described in NRC Generic Letter 93-08, "Relocation of Technical Specification Tables of Instrument Response Time Limits," dated December 29, 1993.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
ORIGINAL SIGNED BY:
William Reckley, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures: 1. Amendment No. 189 to NPF-6
2. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Sincerely,

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2. Safety Evaluation

cc w/encls: See next page

cc:

Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-199

Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street, Slot 30
Little Rock, AR 72205-3867

Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

Manager, Rockville Nuclear Licensing
Framatone Technologies
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 310
London, AR 72847

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

County Judge of Pope County
Pope County Courthouse
Russellville, AR 72801

Vice President, Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 189
License No. NPF-6

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated September 23, 1997, as supplemented on February 27 and March 4, 1998, 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 license 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:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 189, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

William Reckley

William Reckley, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: March 12, 1998

ATTACHMENT TO LICENSE AMENDMENT NO.189

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Revise the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
2-5	2-5
3/4 3-1	3/4 3-1
3/4 3-6	3/4 3-6
3/4 3-6a	-
3/4 3-6b	-
3/4 3-6c	-
3/4 3-10	3/4 3-10
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-19	3/4 3-19
3/4 3-20	3/4 3-20
3/4 6-7	3/4 6-7
B 3/4 3-1	B 3/4 3-1

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
a. Four Reactor Coolant Pumps Operating	≤ 110% of RATED THERMAL POWER	≤ 110.712% of RATED THERMAL POWER
b. Three Reactor Coolant Pumps Operating	*	*
c. Two Reactor Coolant Pumps Operating - Same Loop	*	*
d. Two Reactor Coolant Pumps Operating - Opposite Loops	*	*
3. Logarithmic Power Level - High (1)	≤ 0.75% of RATED THERMAL POWER	≤ 0.819% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2362 psia	≤ 2370.887 psia
5. Pressurizer Pressure - Low	≥ 1717.4 psia (2)	≥ 1686.3 psia (2)
6. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
7. Steam Generator Pressure - Low	≥ 712 psia (3)	≥ 699.6 psia (3)
8. Steam Generator Level - Low	≥ 23% (4)	≥ 22.111% (4)

* These values left blank pending NRC approval of safety analyses for operation with less than four reactor coolant pumps operating.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High	$\leq$21.0 kw/ft (5)	$\leq$21.0 kw/ft (5)
10 DNBR - Low	$\geq$1.25 (5)	$\geq$1.25 (5)
11. Steam Generator Level - High	$\leq$93.7% (4)	$\leq$94.589% (4)

TABLE NOTATION

(1) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\leq 10^{-4}$ of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at $\leq$200 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is $\geq$ 500 psia.

(3) Value may be decreased manually during a planned reduction in steam generator pressure provided the margin between the steam generator pressure and this value is maintained at $\leq$ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and lower level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq 10^{-4}$% of RATED THERMAL POWER.

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE prior to each reactor startup unless performed during the preceding 92 days. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Neutron detectors are exempt from response time testing. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.1.4 The Core Protection Calculator System shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

4.3.1.1.5 The affected Core Protection Calculator Channel shall be subjected to a CHANNEL FUNCTIONAL TEST to verify OPERABILITY within 12 hours of receipt of a valid CPC Cabinet High Temperature alarm.

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TABLE 4.3-1

REACTOR PROTECTION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TESTS	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Linear Power Level - High	S	D(2,4), M(3,4), Q(4)	TA(10)	1, 2
3. Logarithmic Power Level - High	S	R(4)	TA(10), S/U(1)	1, 2, 3, 4, 5 and *
4. Pressurizer Pressure - High	S	R	TA(10)	1, 2
5. Pressurizer Pressure - Low	S	R	TA(10)	1, 2 ,3*,4*,5*
6. Containment Pressure - High	S	R	TA(10)	1, 2
7. Steam Generator Pressure - Low	S	R	TA(10)	1, 2 ,3*,4*,5*
8. Steam Generator Level - Low	S	R	TA(10)	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	TA(10), R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	TA(10), R(6)	1, 2
11. Steam Generator Level - High	S	R	TA(10)	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	TA(10)	1, 2 ,3*,4*,5*
13. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 ,3*,4*,5*
14. Core Protection Calculators	S	D(2,4) R(4,5)	TA(9,10), R(6)	1, 2
15. CEA Calculators	S	R	TA(10), R(6)	1, 2

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensate for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - The CPC CHANNEL FUNCTIONAL TEST shall include the verification that the correct values of addressable constants are installed in each OPERABLE CPC.

(10) - On a STAGGERED TEST BASIS.

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2.1 The Engineered Safety Feature Actuation System (ESFAS) instrumentation channels and bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION requirement of Table 3.3-3 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the Trip Setpoint value.

b. With an ESFAS instrumentation channel inoperable, take the ACTION shown in Table 3.3-3.

SURVEILLANCE REQUIREMENTS

4.3.2.1.1 Each ESFAS instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-2.

4.3.2.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.2.1.3 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ESFAS function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
4. MAIN STEAM AND FEEDWATER ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 712 psia (2)	≥ 699.6 psia (2)
5. CONTAINMENT COOLING (CCAS)		
a. Manual (Trip Buttons)	Not Applicable	Not applicable
b. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
c. Pressurizer Pressure - Low	≥ 1717.4 psia (1)	≥ 1686.3 psia (1)
6. RECIRCULATION (RAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	54,400 ± 2,370 gallons (equivalent to 6.0 ± 0.5% indicated level)	between 51,050 and 58,600 gallons (equivalent to between 5.111% and 6.889% indicated level)
7. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	3120 volts (4)	3120 volts (4)
b. 460 volt Emergency Bus Undervoltage (Degraded Voltage)	423 ± 2.0 volts with an 8.0 ± 0.5 second time delay	423 ± 4.0 volts with an 8.0 ± 0.8 second time delay

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT		TRIP VALUE	ALLOWABLE VALUES
8.	EMERGENCY FEEDWATER (EFAS)		
	a. Manual (Trip Buttons)	Not Applicable	Not Applicable
	b. Steam Generator (A&B) Level-Low	≥ 23% (3)	≥ 22.111% (3)
	c. Steam Generator ΔP-High (SG-A > SG-B)	≤ 90 psi	≤ 99.344 psi
	d. Steam Generator ΔP-High (SG-B > SG-A)	≤ 90 psi	≤ 99.344 psi
	e. Steam Generator (A&B) Pressure - Low	≥ 712 psia (2)	≥ 699.6 psia (2)

(1) Value may be decreased manually, to a minimum of ≥ 100 psia, during a planned reduction in pressurizer pressure, provided the margin between the pressurizer pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is ≥ 500 psia.

(2) Value may be decreased manually during a planned reduction in steam generator pressure, provided the margin between the steam generator pressure and this value is maintained at ≤ 200 psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

(4) Inverse time relay set value, not a trip value. The zero voltage trip will occur in 0.75 ± 0.075 seconds.

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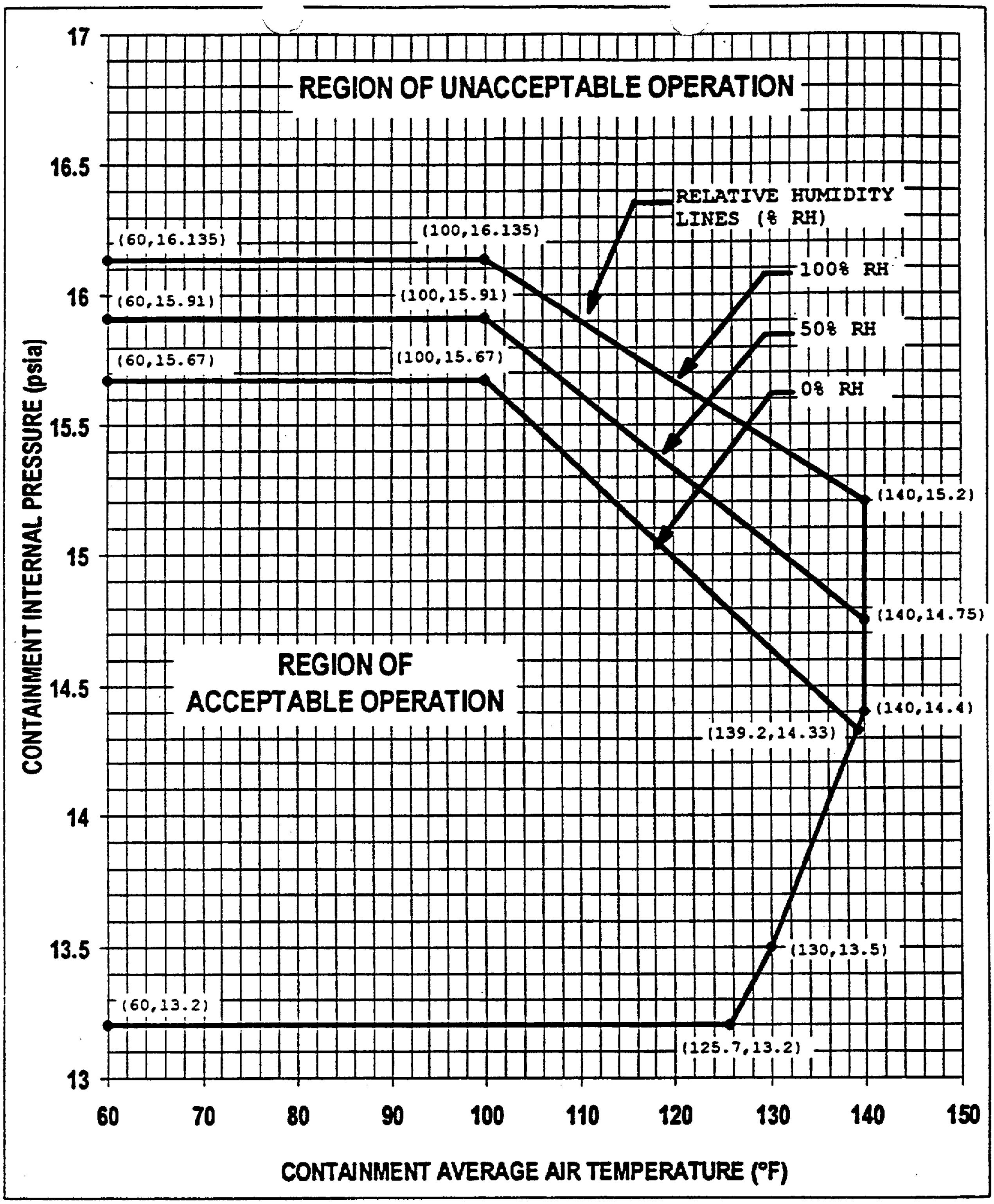

CONTAINMENT INTERNAL PRESSURE vs. AVERAGE AIR TEMPERATURE
FIGURE 3.6-1

NOTE: Instrument Error is not Included on Curve

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.5 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.5.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.5.1 Containment Tendons The containment tendons' structural integrity shall be demonstrated at the end of one, three and five years following the initial containment structural integrity test and at five year intervals thereafter. The tendons' structural integrity shall be demonstrated by a visual examination (to the extent practical and without dismantling load bearing components of the anchorage) of a representative sample* of at least 21 tendons (6 dome, 5 vertical, and 10 hoop) and verifying no abnormal degradation. Unless there is evidence of abnormal degradation of the containment tendons during the first three tests of the tendons, the number of tendons examined during subsequent tests may be reduced to a representative sample of at least 9 tendons (3 dome, 3 vertical and 3 hoop).

*For each inspection, the tendons shall be selected on a random but representative basis so that the sample group will change somewhat for each inspection; however, to develop a history of tendon performance and to correlate the observed data, one tendon from each group (dome, vertical, and hoop) may be kept unchanged after the initial selection.

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The triannual channel functional testing frequency is to be performed on a STAGGERED TEST BASIS.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. The RPS and ESFAS response time tables have been relocated to the Safety Analysis Report (SAR). No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

BASES

Plant Protective System (PPS) logic is designed for operation as a 2-out-of-3 logic, although normally it is operated in a 2-out-of-4 mode.

The RPS Logic consists of everything downstream of the bistable relays and upstream of the Reactor Trip Circuit Breakers. The RPS Logic is divided into two parts, Matrix Logic, and Initiation Logic. Failures of individual bistables and their relays are considered measurement channel failures.

The ESFAS Logic consists of everything downstream of the bistable relays and upstream of the subgroup relays. The ESFAS Logic is divided into three parts, Matrix Logic, Initiation Logic, and Actuation Logic. Failures of individual bistables and their relays are considered measurement channel failures.

Matrix Logic refers to the matrix power supplies, trip channel bypass contacts, and interconnecting matrix wiring between bistable relay cards, up to, but not including the matrix relays. Matrix contacts on the bistable relay cards are excluded from the Matrix Logic definition since they are addressed as part of the measurement channel.

Initiation Logic consists of the trip path power source, matrix relays and their associated contacts, all interconnecting wiring, and the initiation relays (including contacts).

ESFAS Actuation Logic consists of all circuitry housed within the Auxiliary Relay Cabinets (ARCs) used to house the ESF Function; excluding the subgroup relays, and interconnecting wiring to the initiation relay contacts mounted in the PPS cabinet.

For the purposes of this LCO, de-energization of up to three matrix power supplies due to a single failure, such as loss of a vital instrument bus, is to be treated as a single matrix channel failure, providing the affected matrix relays de-energize as designed to produce a half-trip. Although each of the six matrices within an ESFAS Function (e.g., SIAS, MSIS, CSAS, etc.) uses separate power supplies, the matrices for the different ESFAS Functions share power supplies. Thus, failure of a matrix power supply may force entry into the Condition specified for each of the associated ESFAS Functional Units.

Correction letter of 5-10-95

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.189 TO

FACILITY OPERATING LICENSE NO. NPF-6

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated September 23, 1997, Entergy Operations, Inc. (the licensee) submitted a request for changes to the Arkansas Nuclear One, Unit No. 2 (ANO-2) Technical Specifications (TSs). The requested changes would revise the Reactor Protective System (RPS) and Engineering Safety Actuation System (ESFAS) trip set point and allowable values for steam generator low pressure to support continued plant operation with an increased number of plugged steam generator tubes. The proposed amendment would also relocate the RPS and ESFAS response time tables from the TSs to the Safety Analysis Report as described in Nuclear Regulatory Commission (NRC) Generic Letter (GL) 93-08, "Relocation of Technical Specification Tables of Instrument Response Time Limits," dated December 29, 1993.

The letters dated February 27 and March 4, 1998, provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

2.1 Relocation of Instrument Response Time Tables (GL 93-08)

Section 50.36 of Title 10 of the Code of Federal Regulations establishes the regulatory requirements for licensees to include technical specifications as part of applications for operating licenses. The rule requires that technical specifications include items in five specified categories: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The fundamental purpose of the TSs is to impose those conditions or limitations upon reactor operation necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety by identifying those features that are of controlling importance to safety and establishing on them certain conditions of operation which cannot be changed without prior Commission approval. The relocation of requirements, such as instrument response time tables, from the TS to the Updated Safety Analysis Report (USAR), resulted from NRC staff efforts to develop

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improved criteria for delineating those matters that need to be included in TS. The criteria established were included in the final Commission policy statement on TS improvements, published July 22, 1993, (58 FR 39132) and were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36, dated July 19, 1995 (60 FR 36953).

The Commission's final policy statement recognized, as had previous statements related to the staff's TS improvement program, that implementation of the policy would result in the relocation of existing TS requirements to licensee-controlled documents such as the USAR. The NRC issued GL 93-08 and similar line-item TS improvements in order to improve the content and consistency of TSs and to reduce the licensee and staff resources required to process amendments related to those specifications being relocated from the TS to other licensee documents. Those items relocated to the USAR are controlled in accordance with the requirements of 10 CFR 50.59, "Changes, tests and experiments." Section 50.59 of Title 10 of the Code of Federal Regulations provides criteria to determine when facility or operating changes planned by a licensee require prior Commission approval in the form of a license amendment. NRC inspection and enforcement programs also enable the staff to monitor facility changes and licensee adherence to USAR commitments and to take any remedial action that may be appropriate.

2.2 Steam Generator - Low Pressure Setpoint Reduction

ANO-2 has an active damage mechanism affecting the steam generator tubing which requires the repair or the removal of tubes from service when they meet the repair criteria. The unit entered a mid-cycle outage in February 1998, in order to perform inspections of the steam generator tubes and perform plugging of those tubes found to have exceeded the established plugging criteria. A reduction in the heat transfer surface area occurs for each plugged steam generator tube and requires an increased differential temperature across those tubes remaining in service in order to support continued operation at the rated thermal power of the reactor core. The increased differential temperature is achieved by reducing the coolant temperature and steam pressure in the plant's power conversion systems. The lower steam generator pressure anticipated after the mid-cycle outage reduces the operating margin between the full power steam generator pressure and the Plant Protection System (PPS) setpoints based on low steam generator pressure. In order to maintain a comfortable margin between the operating conditions and protection system setpoints, and thereby reduce the occurrence of spurious actuations, the licensee has proposed to reduce the steam generator low pressure setpoints in the TS.

3.0 EVALUATION

3.1 Relocation of Instrument Response Time Tables (GL 93-08)

The licensee has proposed changes to TS 3.3.1.1 and TS 3.3.2.1 that remove the references to Tables 3.3-2 and 3.3-5 and delete these tables from the TS. The licensee has also proposed to relocate TS Figure 3.3-1, "CPC [Core Protection Calculator] Penalty vs. Effective RTD [Resistence Temperature Detector] Time Constant," which is referenced in a footnote to Table 3.3-2. The licensee has relocated the tables and other information related to specific RPS and ESFAS response time limits to the USAR in the USAR update submitted December 9, 1997.

Tables 3.3-2 and 3.3-5 contain the values of the response time limits for the RPS and ESFAS instruments. Figure 3.3-1 provides the values for adjustments to the CPC protection functions if the effective RTD time constant exceeds 8.0 seconds. The limiting conditions for operation for the RPS and ESFAS instrumentation specify these systems shall be operable with the response times as specified in these tables. The limits in Tables 3.3-2 and 3.3-5 are the acceptance criteria for the response time tests performed to satisfy the surveillance requirements of TS 4.3.1.1.3 and TS 4.3.2.1.3 for each applicable RPS and ESFAS trip function. These surveillances ensure that the response times of the RPS and ESFAS instruments are consistent with the assumptions of the safety analyses performed for design basis accidents and transients. The changes associated with the implementation of GL 93-08 involve only the relocation of the RPS and ESFAS response time tables, but retain the surveillance requirement to perform response time testing. The USAR contains the acceptance criteria for the required RPS and ESFAS response time surveillances. Because it does not alter the TS requirements to ensure that the response times of the RPS and ESFAS instruments are within their limits, the staff has concluded that relocation of these response time limit tables from the TS to USAR is acceptable.

The staff's determination is based on the fact that the removal of the specific response time tables does not eliminate the requirements for the licensee to ensure that the protection instrumentation is capable of performing its safety function. Although the tables containing the specific response time requirements are relocated from the TSs to the USAR, the licensee must continue to evaluate any changes to response time requirements in accordance with 10 CFR 50.59. Should the licensee's determination conclude that an unreviewed safety question is involved, due to either (1) an increase in the probability or consequences of accidents or malfunctions of equipment important to safety, (2) the creation of a possibility for an accident or malfunction of a different type than any evaluated previously, or (3) a reduction in the margin of safety, NRC approval and a license amendment would be required prior to implementation of the change.

The staff's review concluded that 10 CFR 50.36 does not require the response time tables to be retained in TSs. Requirements related to the operability, applicability, and surveillance requirements, including performance of testing to ensure response times, for RPS and ESFAS systems are retained due to those systems' importance in mitigating the consequences of an accident. However, the staff determined that the inclusion of specific response time requirements for the various instrumentation channels and components addressed by GL 93-08 was not required. The response times are considered to be an operational detail related to the licensee's safety analyses and are adequately controlled by the requirements of 10 CFR 50.59. Therefore, the continued processing of license amendments related to revisions of the affected instrument or component response times, where the revisions to those requirements do not involve an unreviewed safety question under 10 CFR 50.59, would afford no significant benefit with regard to protecting the public health and safety. The staff has verified that the affected TS requirements have been relocated to the USAR.

In addition to removing the response times from the TS, the licensee is modifying TS Bases Sections 3/4.3.1 and 3/4.3.2 to reflect these changes and to state that RPS and ESFAS response time tables have been relocated to the USAR. These changes are acceptable in that they merely constitute administrative changes required to implement the TS change discussed above.

These TS changes are consistent with the guidance provided in GL 93-08 and the TS requirement of 10 CFR 50.36. The staff has determined that the proposed changes to the TS for ANO-2 are acceptable.

3.2 Steam Generator - Low Pressure Setpoint Reduction

In order to maintain adequate operating margin between the secondary-side steam generator pressures during normal operation and the PPS steam generator low pressure setpoint (related PPS signals initiate a reactor trip, main steam and feedwater isolations, and emergency feedwater actuation), the licensee has proposed to reduce the PPS setpoint from 751 psia to 712 psia. The reduction in the protection system setpoints has the potential to affect the plant's response to those transients that rely on the steam generator low-pressure functions to ensure that safety limits or other design conditions are not exceeded. The licensee has re-analyzed several design-basis transients and accidents to ensure that applicable acceptance limits continue to be satisfied if the steam generator low pressure setpoints are reduced. The main steam line break (MSLB) accident was reanalyzed to determine the effect of the proposed change to the steam generator low pressure setpoint on the limits associated with the reactor core, the peak pressure and temperature within containment, and the radiological consequences to the public and control room personnel.

3.2.1 Reactor Core and Reactor Coolant System Response

The licensee evaluated those Chapter 15 analyses that are potentially affected by the proposed reduction in the steam generator low pressure setpoints. The transients and accidents evaluated are (1) excessive heat removal due to a secondary system malfunction, (2) main steam line break, and (3) main feedwater line break. These analyses also considered a proposed reduction in RCS flow that may be caused by the expected increase in the number of plugged steam generator tubes. A separate TS amendment and safety evaluation (that contains the applicable parts of the following evaluation) are being issued for the reduction in minimum RCS flow rate.

3.2.1.1 Increase in Steam Flow Event

An excess steam demand (ESD) event is caused by a failure of the main steam system that results in an increase in steam flow from the steam generator. In the presence of a negative moderator temperature coefficient, the event results in an increase in core power and a reduction in DNBR. The system response to the event is dependent on the rate of heat transfer through the steam generators. The reduced heat transfer area resulting from steam generator tube plugging will slow down the cooling of the RCS primary system. The reduced initial RCS flow tends to increase the rate of primary cooldown for a given rate of heat transfer. During the transients, the CPC will trip the reactor to avoid violation of DNBR safety limit. To assure that the CPC can accurately sense the cooldown associated with the event, the licensee performed a CPC transient filter analysis for Cycle 13. In the analysis, the limiting conditions (design minimum flow reduced by 10% and no reduction in steam generator heat transfer area) identified by the licensee's sensitivity studies, were assumed for the ESD event. The results show that the minimum acceptable thermal margin to the DNBR limit in the SAR case remains available. Since the results of the existing CPC transient filter analysis verify that CPC trip functions are conservative and demonstrate that the SAR case remains the bounding case, the staff concludes that the effects of a reduction in RCS flow and steam generator tube plugging are appropriately considered for the ESD event.

The licensee also assessed the impact of a lower steam generator low pressure setpoint of 620 psia (reduced from 678 psia in the current SAR analysis) on the ESD event. The event assessed by the licensee is an inadvertent opening of atmospheric dump valves (ADV) event, previously identified as the limiting ESD case. The licensee's assessment shows that a lower steam generator low pressure setpoint delays isolation of the affected steam generator with an opened ADV and results in a 10% increase in the amount of steam release compared to the SAR case. However, the resulting total mass is well within those considered for the main steam line break (MSLB) event. With a greater steam release, the overcooling effect of the MSLB results in limiting core conditions that bound the ESD event. Since the results of MSLB analysis (discussed in section 3.2.1.2) show that the minimum DNBRs are greater than the DNBR safety limit, the licensee stated, and the staff agrees, that the results of the ESD event with a lower steam generator low pressure setpoint can meet the DNBR safety limit. Therefore, the staff concludes that the effects of reduced RCS flow rate and a reduction in the steam generator low pressure setpoint are appropriately considered for the ESD event.

3.2.1.2 Steam Line Break

The licensee reanalyzed the MSLB event with consideration of the effects of a reduction in RCS flow and a decrease in the steam generator low pressure setpoint to close the main steam isolation valves (MSIVs). The analysis was performed with the NRC-approved codes: CENTS for calculations of the system response, ROCS/HERMITE for calculations of the reactivity feedback and peaking factors for hot rods, and HRISE for the DNBR calculations. The licensee used RELAP5/MOD3 to calculate the feedwater flow for the MSLB at hot full power conditions. RELAP5/MOD3 is not an approved code for licensing calculations. At the staff's request, the licensee provided the feedwater flow rates calculated with the RELAP5/MOD3 code and compared them with the flow rates calculated with the HSTA code, a code used in the approved MSLB analysis to support the licensing amendments for ANO-2. The comparison shows that RELAP5/MOD3 predicts a higher flow rate throughout the transient. The use of higher feedwater flow rates increases the overcooling effects and is conservative. The staff has determined that the use, in the SLB analysis, of feedwater flow rates that are higher than those calculated by HSTA is conservative and is therefore acceptable. The SLB analysis is, therefore, adequate and acceptable for ANO-2. The staff notes, however, that this action does not approve the use of RELAP5/MOD3 computer code for this or any other licensing analysis for ANO-2. Future use of REPAP5/MOD3 for licensing applications should be preceded by staff review and approval of the code and the its specific application.

The licensee performed analyses for 4 double-ended guillotine MSLB (with break sizes of 6.357 ft^2) cases in order to determine the limiting cases for approaching the fuel design limits. The 4 cases analyzed are:

1. A large MSLB during full power (HFP) conditions in combination with a single failure, loss of offsite power and a stuck CEA.

2. Case 1 with offsite power available.

3. A large MSLB during zero power (HZP) conditions in combination with a single failure, loss of offsite power and a stuck CEA.

4. Case 3 with offsite power available.

To maximize the overcooling effect, the licensee made the following assumptions: (1) the highest actuation pressure for a safety injection actuation signal (SIAS) was assumed to delay the injection borated water to the RCS, (2) the cooldown of the RCS was terminated when the affected steam generator blew dry, (3) a CPC low DNBR trip was credited for the loss of offsite power cases and the setpoint was based on 96.5% of the RCP design speed, (4) a low steam generator pressure was assumed at 620 psia to trip the reactor and to actuate the steam generator low pressure signal that closed the main steam isolation valves (MSIVs), main feedwater isolation valves (MFIVs), and back-up MFIVs, (5) the most negative moderator temperature and Doppler coefficients were used to maximize the reactivity addition resulting from the cooldown effect, (6) two emergency feedwater pumps were assumed to be available to maximize the cooling potential of the EFW system, and (7) the boron from the safety injection tanks was not credited.

For single failure considerations, the analyses assumed that for the loss of offsite ac power cases, one emergency diesel generator (EDG) failed to start. The failure of an EDG resulted in the failure of one high-pressure safety injection (HPSI) pump and the MFIVs to close. For the HFP case with ac power available, a bus fast transfer failure was identified as the worst single failure. The single failure resulted in the failure of the back-up MFIVs and a HPSI pump. For the HZP case with ac power available, a single failure of a HPSI train was assumed.

The analyses show that the HFP cases remain subcritical throughout the post trip event and that the HZP cases show a return-to-criticality that is bounded by the SAR results. The calculated DNBRs for all cases are greater than the DNBR safety limit and, thus ensure that no fuel failure will occur. Since the licensee used NRC-approved codes for analyses, the values used for input parameters are conservative, and the results show that the minimum calculated DNBRs are greater than the acceptable safety limit to assure fuel integrity, the staff concludes that the analyses are acceptable.

3.2.1.3 Feedwater Line Break

The licensee performed sensitivity studies of a 10% reduction in the RCS flow and 30% steam generator tube plugging on the feedwater line break (FLB) analysis presented in the SAR. The results show that changes in initial RCS flow have minimal effects on the FLB analysis, and that the cases without assumed steam generator tube plugging result in a slightly higher peak RCS pressure. Since a minimum design RCS flow rate without steam generator tube plugging are assumed in the SAR case, the licensee's sensitivity studies demonstrates that the SAR case remains conservative for the FLB analysis.

To assess the effect of a lower steam generator low pressure setpoint (620 psia) to close the MSIVs during the events, the licensee reanalyzed the feedwater line break (FLB) event with loss of ac power, which is the limiting case identified in the SAR.

The licensee performed FLB analyses for various break sizes with the approved CENTS code and identified that the break of 0.24 ft^2 resulted in the highest peak RCS pressure. To maximize the calculated peak RCS pressure, the licensee made the following assumptions: (1) the least negative Doppler coefficient corresponding to the BOC core was used to maximize the power increase, (2) the initial plant conditions were assumed to be during full power operation with a loss of offsite power at the time the reactor trip breakers open, (3) a conservative CEA insertion curve corresponding to the axial power shape of +0.6 ASI was assumed, (4) a steam generator low pressure signal was assumed at 620 psia to actuate the MSIVs with a closure time of 3 seconds, (5) the blowdown of saturated liquid from the affected steam generator was assumed, (6) the tolerance for the safety valves and secondary safety valves was assumed to be +3% of the setpoints, and (7) the decay heat was maximized by assuming an equilibrium core.

The initial pressure and initial steam generator inventories were selected such that the low steam generator water level trip in the intact steam generator and the high pressurizer pressure trip occurred simultaneously with the dryout of the affected steam generator. The sensitivity study showed that this assumption resulted in a maximum peak RCS pressure after the trip.

The results of the reanalyses show that the peak RCS pressure is 2730.1 psia which is less than 110% of the design pressure. Since the licensee used NRC-approved codes for the analysis, the values used for input parameters are conservative, and the results show that the peak calculated RCS pressure is within the acceptance criteria of 110% of the design pressure, the staff concludes that the reanalyses are acceptable.

3.2.1.4 Conformance to SER Conditions

As a result of the findings from the NRC's Maine Yankee Lessons Learned Task Force, the staff requested that the licensee verify conformance to conditions stated in NRC safety evaluation reports (SERs) associated with the topical reports (TRs) reviewed and approved by the NRC staff for referencing in licensing applications. In response to the staff's request, the licensee and the fuel vendor, ABB-CE, evaluated their compliance with the conditions specified in the staff's SERs for those topical reports referenced in the submittals pertaining to the proposed TS changes (reduction in minimum RCS flow rate and reduction in the steam generator low pressure setpoint). The licensee's evaluation determined that the SER conditions associated with the referenced topical reports had been satisfied. Accordingly, the staff finds that the licensee has adequately addressed the request to ensure that previously approved topical reports are used consistent with the limitations established as a condition of NRC approval.

3.2.2 Containment Response

A lower steam generator pressure setpoint may delay the receipt of reactor trip and main steam and feedwater isolation signals following a main steam line break. Such delays would result in an increase in the mass and energy released into the containment following a MSLB. The licensee analyzed a spectrum of break sizes, power levels, and initial conditions within the containment building. Based upon the time at which the setpoint would be reached, the licensee assumed conservative setpoints that accounted for instrument uncertainties for either normal conditions or harsh environments. The licensee calculated mass/energy releases into the containment using a combination of the RELAP5MOD3.1 for feedwater flow/enthalpy data and the SGN-III code for blowdown mass and energy values. The SGN-III output was input to the COPPATTA containment code to predict the containment response to the MSLB.

The analysis techniques and assumptions are generally the same as those currently described in the USAR. A study performed as part of this analysis determined that the limiting single failure related to the containment response to a MSLB is the temporary loss of a vital electrical bus due to failure of the fast transfer mechanisms. This failure results in delays in (1) a containment fan cooler to start, (2) a containment spray pump to start, and (3) the closure of the back-up main feedwater isolation valves, until startup of the associated emergency diesel generator. Based on the mass and energy data from SGN-III and the containment modeling within the COPPATTA code, the peak containment conditions following a MSLB were determined to be 53.0 psig and 423 °F. These values are less than the containment design pressure of 54 psig and the current MSLB peak temperature estimate of 426 °F.

Following the analysis of the various MSLB scenarios for different break sizes and initial conditions, an error was discovered in the assumed maximum break size. The maximum pipe break area had been assumed to be 6.19 ft^2 which is less than the maximum installed area given a guillotine break in the main steam line (6.357 ft^2). The analysis was repeated with the larger break area considering the limiting single failure, an increase in the assumed service water temperature to 120 °F to be consistent with current USAR assumption, and a correction (addition) of heat sink data. In order to offset the increase in the mass and energy release due to the larger break area, a steam generator tube plugging level of 10% was assumed in the revised analysis (resulting in a lower initial steam pressure and reduced heat transfer from the reactor coolant system during the MSLB accident). Given that the plugging level in the ANO-2 steam generators currently exceeds 13%, the staff finds that the 10% assumption is conservative and can be used to offset the needed changes in break area and service water temperature. Changes were also assumed in the response time of the containment spray trains by assuming an earlier delivery of spray to the containment than was assumed in the 6.19 ft^2 case. The revised assumptions of a faster containment spray response removed margin that the licensee had added for future considerations but remained bounded by the response times currently in TS Table 3.3-5. The revised analysis resulted in a peak containment pressure of 52.3 psig which remains below the containment design pressure of 54.0 psig. The staff finds that the licensee has adequately demonstrated that the proposed reduction in the steam generator - low pressure setpoint will not result in exceeding the design limits of the containment following the rupture of a main steam line.

TS Figure 3.6-1, "Containment Internal Pressure vs. Average Air Temperature," is developed using the limiting initial conditions for analyses associated with (1) the loss of coolant accident evaluation performed per 10 CFR 50.46, (2) the containment design negative pressure differential of 5psid (potentially caused by an inadvertent actuation of containment spray), and (3) the containment design pressure of 54 psig following either a loss of coolant accident or secondary-side high energy line break. The licensee presented the results of a series of analyses related to containment initial conditions (pressure, temperature, and relative humidity) for the revised main steam line break conditions. The new analyses, combined with previous evaluations of the sensitivity of the containment pressure response to initial conditions for the loss of coolant accident and the limiting initial conditions for the loss of coolant evaluation model and negative pressure differential limits, resulted in minor changes to the locus of points defining the area of acceptable operation in Figure 3.6-1. The staff finds the changes to the figure are consistent with the analyses presented by the licensee and are acceptable.

3.2.3 Assessment of Radiological Consequences

The licensee performed an assessment of the radiological dose consequences of a MSLB accident in support of the proposed change in the steam generator low pressure setpoints. That assessment was based upon a primary to secondary leakage of 1.0 gpm (300 gpd) allowed by TS 3.4.6.2. The licensee assumed that the 1.00 gpm leakage was divided into 0.5 gpm to the faulted steam generator and 0.5 to the intact steam generators. The licensee found the radiological dose consequences acceptable, assuming allowable activity levels in the primary coolant of 60 μCi/g dose equivalent ^{131}I for a pre-existing spike condition and 1.0 μCi/g dose equivalent ^{131}I for the accident-initiated spike condition.

The staff has independently calculated the doses resulting from a MSLB accident using the methodology in Standard Review Plan (SRP) 15.1.5, Appendix A. The staff performed two separate assessments. The first assessment was based upon a pre-existing iodine spike activity level of 60 μCi/g of dose equivalent ^{131}I in the primary coolant. The second assessment was based upon an accident-initiated iodine spike. Both assessments utilized dose conversion factors listed in TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." For the accident initiated spike assessment, the staff assumed that the accident initiated an increase in the release rate of iodine from the fuel by a factor of 500 over the release rate to maintain an activity level of 1 μCi/g of dose equivalent ^{131}I in the primary coolant.

For each assessment, the staff calculated doses for individuals located at the Exclusion Area Boundary (EAB) and at the Low-Population Zone (LPZ). The control room operator's thyroid dose was also calculated. The staff also reviewed the licensee's description of the revised MSLB accident analysis and the postulated dose results. The results of the staff's independent calculations described above were used to confirm the acceptability of the licensee's analysis methodology. Based on comparisons of results, the staff found the licensee's analysis to be appropriate.

The staff has concluded, based upon the considerations above, that the proposed change to the Technical Specifications is acceptable. The staff has determined that reasonable assurance exists that, in the event of a postulated MSLB, the doses to persons at the EAB and LPZ would continue to be well within 10 CFR Part 100 dose guidelines, and that the postulated control room operator doses would continue to be less than the criteria in the SRP and 10 CFR Part 50, Appendix A, GDC 19.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 4311). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: S. Sun, NRR/SRXB
M. Blumberg, NRR/PERB
W. Reckley, NRR/PD41

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