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

December 31, 1998

SUBJECT: ISSUANCE OF AMENDMENT NO.197 TO FACILITY OPERATING LICENSE NO. NPF-6 - ARKANSAS NUCLEAR ONE, UNIT NO. 2 (TAC NO. MA2222)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment No. 197 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated June 29, 1998 (2CAN069804), as supplemented by letter dated December 17, 1998 (2CAN129804), and December 22, 1998 (2CAN129805).

This amendment revises the as-found lift setting tolerance for the ANO-2 main steam safety valves and the pressurizer safety valves, revises the maximum allowable linear power level-high trip setpoint with inoperable steam line safety valves, and relocates part of the specifications for steam line safety valves to the ANO-2 Safety Analysis Report. Administrative and bases changes have also been made.

A copy of the 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:

M. Christopher Nolan, 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. 197 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

December 31, 1998

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This amendment revises the as-found lift setting tolerance for the ANO-2 main steam safety valves and the pressurizer safety valves, revises the maximum allowable linear power level-high trip setpoint with inoperable steam line safety valves, and relocates part of the specifications for steam line safety valves to the ANO-2 Safety Analysis Report. Administrative and bases changes have also been made.

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cc w/encls: See next page

Mr. C. Randy Hutchinson
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 2

cc:

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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. 197
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 June 29, 1998, as supplemented by letters dated December 17, 1998, and December 22, 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 1;

   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. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 197, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

M. Christopher Nolan

M. Christopher Nolan, 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: December 31, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 197

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace 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 pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4
3/4 7-2	3/4 7-2
3/4 7-4	3/4 7-4
B 3/4 7-1	B 3/4 7-1
6-21a	6-21a

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA ± 3%*#.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a ± 1% tolerance band, the setting shall be adjusted to within ± 1% of the lift setting shown.

#Equivalent relief capacity during MODE 5 satisfies the requirements of this Specification. This allows both pressurizer code safety valves to be removed for testing and/or maintenance.

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting 2500 psia ± 3%*.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

b. The provisions of specification 3.0.4 may be suspended for one valve at a time for up to 18 hours for entry into and during operation in MODE 3 for the purpose of setting the pressurizer code safety valves under ambient (hot) conditions provided a preliminary cold setting was made prior to heatup.

SURVEILLANCE REQUIREMENTS

4.4.3 No additional Surveillance Requirements other than those required by Specification 4.0.5.

*The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a ± 1% tolerance band, the setting shall be adjusted to within ± 1% of the lift setting shown.

TABLE 3.7-1

MAXIMUM ALLOWABLE LINEAR POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Linear Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	91.0
2	67.7
3	36.0

TABLE 3.7-5

STEAM LINE SAFETY VALVES

	VALVE NUMBER		LIFT SETTING (± 3%) *
	Line No. 1	Line No. 2	
a.	2PSV-1002	2PSV-1052	1078 psig
b.	2PSV-1003	2PSV-1053	1105 psig
c.	2PSV-1004	2PSV-1054	1105 psig
d.	2PSV-1005	2PSV-1055	1132 psig
e.	2PSV-1006	2PSV-1056	1132 psig

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a ± 1% tolerance band, the setting shall be adjusted to within ± 1% of the lift setting shown.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY CYCLES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% of its design pressure of 1100 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The "as-found" requirements are consistent with Section XI of the ASME Boiler and Pressure Vessel Code, 1986 Edition, and Addenda through 1987. The MSSV rated capacity passes the full steam flow at 102% RATED THERMAL POWER (100% + 2% for instrument error) with the valves fully open. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoints are determined from the most conservative value obtained from two methods. The first method calculates setpoints based on the reduction in relieving capacity with the number of MSSVs inoperable and are derived as follows:

$$SP = \frac{(X) - (Y)}{X} \times (Z\%)$$

where:

SP = Reduced reactor trip setpoint in percent of RATED THERMAL POWER

Z% = Total maximum safety valve relieving capacity of 102% power steam flow

X = Total relieving capacity of all safety valves per steam line in lbs/hour

Y = Total maximum relieving capacity of the inoperable safety valve(s) in lbs/hour. In each case, the valves with the greatest relieving capacity were assumed inoperable.

In the second method, the setpoint is determined from the Loss of Condenser Vacuum (LOCV) event. With + 3% MSSV tolerance, the LOCV event is analyzed with 1, 2, or 3 inoperable MSSVs on any steam generator, to determine the allowable initial power levels (and the trip setpoints) that yield acceptable results. For each of the inoperable MSSV conditions (1, 2, or 3 inoperable) in Table 3.7-1, the more conservative trip setpoint from the two analytical methods is selected.

CORE OPERATING LIMITS REPORT

10) "Calculative Methods for the CE Small Break LOCA Evaluation Model," CENPD-137, Supplement 2-P-A, dated April, 1998 (Methodology for Specification 3.1.1.4 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

11) "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating CEA and Group P Insertion Limits, and 3.2.4.b for DNBR Margin).

12) "Technical Manual for the CENTS Code," CENPD 282-P-A, February 1991 (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margin, 3.1.1.4 for MTC, 3.1.3.1 for CEA Position, 3.1.3.6 for Regulating and Group P Insertion Limits, and 3.2.4.b for DNBR Margin.

13) Letter: O.D. Parr (NRC) to F.M. Stern (CE), dated June 13, 1975 (NRC Staff Review of the Combustion Engineering ECCS Evaluation Model). NRC approval for 6.9.5.1.4, 6.9.5.1.5, and 6.9.5.1.8 methodologies.

14) Letter: O.D. Parr (NRC) to A.E. Scherer (CE), dated December 9, 1975 (NRC Staff Review of the Proposed Combustion Engineering ECCS Evaluation Model changes). NRC approval for 6.9.5.1.6 methodology.

15) Letter: K. Kniel (NRC) to A.E. Scherer (CE), dated September 27, 1977 (Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P). NRC approval for 6.9.5.1.9 methodology.

16) Letter: 2CNA038403, dated March 20, 1984, J.R. Miller (NRC) to J.M. Griffin (AP&L), "CESEC Code Verification." NRC approval for 6.9.5.1.11 methodology.

6.9.5.2 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.5.3 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 197TO

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 June 29, 1998 (2CAN069804), as supplemented by letters dated December 17, 1998 (2CAN129804), and December 22, 1998 (2CAN129805), Entergy Operations, Inc. (EOI or the licensee), submitted a request for changes to the Arkansas Nuclear One, Unit No. 2, Technical Specifications (TSs).

The proposed amendment would revise the as-found lift setting tolerance for the ANO-2 main steam safety valves and the pressurizer safety valves, revise the maximum allowable linear power level-high trip setpoint with inoperable steam line safety valves, and relocate part of the specifications for steam line safety valves to the ANO-2 Safety Analysis Report (SAR). Administrative and bases changes would also be made.

The information in the December 17, 1998 (2CAN129804), and December 22, 1998 (2CAN129805), submittals provided clarifying information and did not expand the scope of the original application as initially noticed, or change the staff's proposed no significant hazards determination published in the Federal Register on October 21, 1998 (63 FR 56242).

2.0 BACKGROUND

The reactor coolant system (RCS) has two pressurizer safety valves (PSVs) to provide overpressure protection during normal power operations (other valves provide protection during low temperature operations). They are direct acting, spring-loaded valves meeting ASME Code requirements. As stated in SAR Section 5.5.13, "Reactor Coolant System, Component and Subsystem Design, Safety and Relief Valves," these valves are designed to pass sufficient steam to limit the RCS pressure to 110 percent of the 2500 psia design pressure following a complete loss of turbine generator load without a simultaneous reactor trip.

Overpressure protection for the shell side of the steam generators and portions of the main steam lines is provided by 10 direct acting, spring-loaded ASME Code safety valves (main steam safety valves or MSSVs). As stated in SAR Section 10.3.2, "Steam and Power

Conversion System, Main Steam Supply System, System Description," these valves can pass a steam flow equivalent to a power level of 2900 MWt at the nominal set pressure. With one or more of these valves inoperable, TSs (see TS 3/4.7.1, "Plant Systems - Turbine Cycle - Safety Valves," and TS Table 3.7-1) require that reactor power be reduced, and that the maximum allowable linear power level-high trip setpoint be reduced.

3.0 EVALUATION

Each TS change proposed by the licensee is discussed below.

3.1 TS 3.4.2, "Reactor Coolant System - Safety Valves - Shutdown" and TS 3.4.3, "Reactor Coolant System - Safety Valves - Operating"

TS 3.4.2 currently reads

> A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA +1, -3%*#.

The licensee proposes to change the upper bound, so that TS 3.4.2 would read

> A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA +/- 3%*#.

TS 3.4.3 currently reads

> All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 PSIA +1, -3%*.

The licensee proposes to change the upper bound, so that TS 3.4.3 would read

> All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 PSIA +/- 3%*.

The "*" footnote for each TS reads

> * The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a +/- 1% tolerance band, the setting shall be adjusted to within +/- 1% of the lift setting shown.

and would not be changed.

TS 4.0.5 (surveillance requirements for inservice inspection and testing of ASME Code Class 1, 2 and 3 components) specifies that the PSV and MSSV testing must conform to the ASME Boiler and Pressure Vessel Code and applicable addenda. The Code (Section XI, 1986 Edition, and Addenda through 1987) allows for as-found PSV and MSSV lift setting tolerances of +/- 3% when the licensee can demonstrate that the valves will still perform their safety function at these limits.

Increasing the upper bound of the lift setting tolerance of the PSVs (during operations and while shutdown) from +1% to +3%, with the caveat that this increase would be for as-found conditions only, will allow normal surveillance testing of the PSVs (with no reporting requirements) to be within +3% of the nominal lift setpoint of 2500 psia.

The licensee performed the analyses to justify the proposed changes. EOI stated that a feedwater line break is the accident which most significantly challenges the PSVs. The licensee's current calculated peak pressure for this event is 2742 psia (SAR Table 15.1.14-20), which is acceptable because it does not exceed 110% of the design value of 2500 psia (that is, 2750 psia). This event was reanalyzed for Cycle 13 operation (and bounds the upcoming Cycle 14) for several different steam generator tube plugging levels, and accounted for an actual PSV lift setting of +3% of nominal. The assessment was performed using the *Combustion Engineering Nuclear Transient Simulation* (CENTS), which is an NRC-approved methodology. The licensee confirmed that all the conditions of the NRC SER approving the methodology were satisfied. Additionally, the licensee utilized appropriate conservative assumptions that were originally reviewed and approved by the NRC staff in the issuance of Amendment Nos. 189 and 190. The licensee determined that the peak RCS pressure would be 2730.1 psia.

The licensee stated that the limiting peak RCS pressure anticipated operational occurrence (AOO) is the loss-of-condenser-vacuum (LOCV) event. The licensee's current calculated peak pressure for this event is 2671 psia (SAR Table 15.1.7-2), which is acceptable because it does not exceed 110% of the design value of 2500 psia, which is 2750 psia. This event was also reanalyzed using the CENTS simulation for different tube plugging levels, and included the proposed +3% lift setting tolerance. The results of this assessment, which utilized conservative assumptions and for which the licensee confirmed that all conditions of the NRC SER were satisfied, was a predicted peak pressure of 2683 psia, which is lower than the predicted pressure for the feedwater line break accident, and hence this AOO does not result in the limiting pressure for the PSV lift setting tolerance change.

Since the calculated peak pressure due to the limiting event (feedwater line break accident) does not exceed 110% of the design value of 2500 psia (2750 psia), and since the calculation was performed using an NRC-approved code with appropriate conservative assumptions as determined by the staff in the approval of Amendment Nos. 189 and 190, and since the higher lift setting tolerance satisfies the ASME code requirements, and since the as-left setting tolerance will remain +/- 1% of nominal, the increase in the as-found tolerance for the PSV lift pressure to +3% is acceptable for TS 3.4.2 and TS 3.4.3.

3.2 <u>TS Table 3.7-5, "Steam Line Safety Valves" - Lift Setting Tolerance Change</u>

TS Table 3.7-5 currently specifies the steam line safety valve lift setting upper bound as +1% of nominal pressure. The licensee proposes to change the upper bound to +3% of nominal. Additional information is provided in a footnote which reads:

> * The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure. If found outside of a +/- 1% tolerance band, the setting shall be adjusted to within +/- 1% of the lift setting shown.

(This is identical to the PSV lift setting footnote) and would not be changed. Increasing the upper bound of the lift setting tolerance of the MSSVs from +1% to +3%, with the caveat that this increase would be for as-found conditions only, will allow normal surveillance testing of the MSSVs (with no reporting requirements) to be within +3% of the nominal lift setpoint of 2500 psia.

The licensee stated that a feedwater line break is also the accident which most significantly challenges the MSSVs. The licensee's current calculated peak pressure for this event is approximately 1100 psia (SAR Figure 15.1.8-3), which is acceptable because it does not exceed 110% of the design value of 1100 psia (that is, 1210 psia). This event was reanalyzed for Cycle 13 operation (and bounds the upcoming Cycle 14), and accounted for actual MSSV lift settings of +3% of nominal. The assessment was performed using the CENTS methodology, and utilized appropriate conservative assumptions. In addition, EOI confirmed that all of the SER conditions were satisfied. The licensee determined that the peak secondary pressure would be 1163.8 psia.

The licensee stated that the limiting peak RCS pressure AOO is the LOCV event. The licensee's current calculated peak secondary pressure for this event is 1144 psia (SAR Table 15.1.7-2), which is acceptable because it does not exceed 110% of the design value of 1100 psia (that is, 1210 psia). This event was also reanalyzed using the CENTS simulation, including the proposed +3% lift setting tolerance. The results of this assessment, which utilized conservative assumptions as determined by the staff with the approval of Amendments Nos. 189 and 190 and for which the licensee confirmed that all of the SER conditions were satisfied, was a predicted peak secondary pressure of 1195 psia, which is higher than the predicted pressure for the feedwater line break accident, and hence this AOO leads to the limiting secondary pressure for the MSSV lift setting tolerance change.

The increase in the MSSV as-found lift pressure impacts the peak clad temperature (PCT), and consequently the cladding oxidation, during a small-break loss-of-coolant accident (SBLOCA). The licensee determined that the increase in MSSV lift pressure results in a higher steam generator (SG) pressure, which results in a higher RCS pressure during a SBLOCA. The higher RCS pressure leads to decreased safety injection flow and increases the break flow, resulting in a higher PCT.

To ensure that the criteria of 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems [ECCSs] for Light-Water Nuclear Power Plants," continue to be satisfied by the ANO-2 ECCSs with higher lift-setting MSSV tolerances, the SBLOCA was reanalyzed using the NRC-approved methodology described in CENPD-137, Supplement 2-P-A, *Calculative Methods for the ABB CE Small Break Evaluation Model* (S2M). The licensee confirmed that there are no conditions in the NRC SER approving the methodology that need to be specifically addressed. The licensee provided a comparison between the S1M and S2M models in their November 24, 1996 (2CAN119610) submittal for Amendment No. 179. This submittal contained an analysis which demonstrated the sensitivity of these codes to break size and steam generator pressure. This information was used in determining the limiting break size of 0.05 ft^2 as described in SAR Section No. 6.3.3.2.3.5 and demonstrated that the 0.05 ft^2 break size remained bounding for both the S1M and S2M models. Since this sensitivity study was performed using the +3% main steam safety valve lift setting tolerance as stated in the licensee's letter dated December 22, 1998 (2CAN129805), the results of this analysis are appropriate for the identification of the limiting break size. The licensee also indicated in their December 22, 1998 (2CAN129805) letter that the same inputs and assumptions associated with this previous analysis were conservative and remain applicable for the proposed change.

The results of the calculated parameters used to satisfy criteria 1, 2, and 3 of 10 CFR 50.46 are summarized below:

Parameter	Criterion	Current Value	New Value
Peak Clad Temperature (degrees Fahrenheit)	< 2200	2011	1798
Maximum Cladding Oxidation (percent)	< 17	5.47	4.8
Core-Wide Cladding Oxidation (percent)	< 1	< 0.835	< 0.36

Though the increase in MSSV lift pressure results in a higher steam generator (SG) pressure, which results in a higher RCS pressure during a SBLOCA, the new calculated PCT is actually lower than the current value due to the differences in the calculation methodologies. Based on their analysis as summarized by the information provided above, the licensee has concluded that the proposed change satisfies criteria 4 and 5 of 10 CFR 50.46. The NRC staff has reviewed the information provided and concluded that the proposed change meets the requirements of 10 CFR 50.46.

Since the calculated peak pressure due to the limiting event (LOCV AOO) does not exceed 110% of the design value of 1100 psia (1210 psia), and since the calculation was performed using an NRC-approved code with appropriate conservative assumptions, and since the higher lift setting tolerance satisfies the ASME code requirements, and since the criteria of 10 CFR 50.46 continue to be satisfied, and since the as-left setting tolerance will remain +/- 1% of nominal, the increase in the as-found tolerance for the MSSV lift pressure to +3% is acceptable. Future changes to input parameters which affect this analysis will be handled through the core operating limits report (COLR) process which ensures the appropriate configuration controls and regulatory processes are utilized.

3.3 TS Table 3.7-5, "Steam Line Safety Valves" - Removal of Orifice Size

TS Table 3.7-5 currently specifies the orifice size of each MSSV. The licensee noted that this information is already contained in the SAR, and proposes to remove this information from the TS table.

Criteria for inclusion of requirements in the TSs is provided in 10 CFR 50.36, as follows:

Criterion 1:

> Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2:

> A process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3:

> A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a Design Basis Accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4:

> A structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

Requirements that are in the existing TSs, but do not meet the criteria set forth in 10 CFR 50.36 for inclusion in TS, can be relocated to appropriate licensee-controlled documents.

The MSSVs are part of the primary success path to mitigate accidents and transients as described in Criterion 3. The lift settings of the MSSVs are critical to the proper functioning of the safety valve, and will be maintained in the TSs. Verification of the safety valve lift setting provides assurance that the valve will function as appropriate to mitigate a pressure transient. The orifice size is a sub-component design detail that does not meet the criteria established in 10 CFR 50.36 for the inclusion of requirements in the Technical Specifications. The orifice is a passive sub-component which controls the maximum relief capacity of the safety valve after the valve has fully lifted and therefore does not satisfy Criterion 3. Criterion 1, 2, and 4 do not apply to the MSSVs as the safety valve does not provide indication, does not establish the initial conditions for a design basis accident or transient, and has not shown to be significant to public health and safety. The orifice size is listed in the SAR along with other design information associated with the MSSVs. The SAR is the appropriate licensing document to contain the design detail describing the technical attributes of a component that support the execution of its design function. The staff has determined that, since the orifice size does not meet the criteria described above, removal of the orifice size from TS Table 3.7-5 is consistent with the requirements of 10 CFR 50.36. The description of the orifice size in the SAR ensures that the process specified in 10 CFR 50.59 will provide the appropriate controls for future changes. Therefore, the proposed removal is acceptable.

3.4 TS Table 3.7-1, "Maximum Allowable Linear Power Level-High Trip Setpoint With Inoperable Steam Line Safety Valves During Operation With Both Steam Generators"

The maximum allowable linear power level-high trip setpoint values provided in this table are currently determined using a calculation based on MSSV relieving capacity, as stated in TS Bases 3/4.7.1.1. The licensee is proposing to include a second calculation method based on the analysis of the LOCV event (assuming +/- 3% lift setting tolerance). This AOO leads to the

limiting secondary pressure. In addition, the licensee proposes to utilize the more conservative trip setpoint from the two methods.

The LOCV event was analyzed using the NRC-approved CENTS methodology which is appropriate for use at ANO. The licensee confirmed that the SER conditions associated with the approval of this methodology have been satisfied. Since an approved code was used and the SER conditions were satisfied, the use of this second methodology is acceptable. In addition, the continued use of the current methodology together with the new methodology is acceptable because the most conservative setpoint is being utilized. Therefore, the proposed change is acceptable.

3.5 TS 6.9.5.1, "Administrative Control - Core Operating Limits Report," Analytical Methods

This TS provides a list of the NRC-approved analytical methods used to determine core operating limits. The licensee proposes to add a reference to "Calculative Methods for the CE Small Break LOCA Evaluation Model" (Topical Report CENPD-137, Supplement 2-P-A, dated April 1998) and to renumber the other references as appropriate.

Since this methodology has been approved by the NRC, is appropriate for use at ANO, and will ensure that values for cycle specific parameters will be determined such that applicable limits of the plant safety analysis are met, the staff finds it acceptable to reference in the TSs for use in the COLR process.

3.6 TS Bases 3/4.7.1.1, "Plant Systems - Bases - Turbine Cycle - Safety Cycles"

The licensee has proposed changes to this bases section related to the MSSV lift settings, reactor trip settings and setpoint methodology. Since these changes are consistent with the TSs, they are acceptable. In addition, the licensee is removing relief valve capacity information from the TS Bases. The staff has concluded that this change is acceptable since it is consistent with NUREG-1432 and this information is already captured in the SAR.

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

This 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 or changes a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 56242). This amendment also changes reporting or record keeping requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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 staff 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Date: December 31, 1998