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

September 29, 2000

Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGES AND UNREVIEWED SAFETY QUESTION RESOLUTION RELATED TO APPLICABLE LIMITS AND SETPOINTS FOR STEAM GENERATOR REPLACEMENT (TAC NO. MA7299)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 222 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit 2 (ANO-2). This amendment consists of changes to the License and Technical Specifications (TSs) in response to your application dated November 29, 1999, as supplemented by letters dated January 26, May 17 (2 letters), May 31, and August 4, 2000.

The amendment revises the License and TSs to maintain consistency with the transient and accident analyses which evaluated the impact of the replacement steam generators (SGs) that are being used for Cycle 15 operation. The License is changed to incorporate a new methodology employed in calculating radiological doses for some non-loss-of-coolant accident events. TS changes are made to the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) low pressurizer pressure setpoints, the RPS and ESFAS low SG pressure setpoints, the RPS and ESFAS low SG level setpoints, the reactor coolant flow rate limit, and the high linear power trip setpoints with inoperable main steam safety valves (MSSVs).

The amendment also makes changes to the TSs that are not directly related to the replacement SGs. These changes revise the allowed outage time of the MSSVs in Modes 1 and 2 to allow up to 12 hours to reduce the high linear power level-high trip setpoint when one or more MSSVs are inoperable, and revise the action statement in Mode 3 to maintain at least two MSSVs operable on each SG.

Changes to the Bases that correspond to the above License and TS changes are being incorporated with the amendment.

DF01

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

Sincerely,

/RA/

Thomas W. Alexion, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-368

Enclosures:
1. Amendment No. 222 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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 222
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 November 29, 1999, as supplemented by letters dated January 26, May 17 (2 letters), May 31, and August 4, 2000, 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, by Amendment No. 222, the Facility Operating License No. NPF-6 is amended to incorporate a new methodology employed in calculating radiological doses as set forth in the application for amendment by Entergy Operations, Inc. dated November 29, 1999, as supplemented by letters dated January 26, May 17 (2 letters), May 31, and August 4, 2000, and evaluated in the staff's safety evaluation enclosed with this amendment.

3. Accordingly, the license is also 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. 222, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

4. The license amendment is effective as of its date of issuance and shall be implemented prior to startup from the 2R14 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 29, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 222

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
2-5	2-5
B 2-4	B 2-4
3/4 2-7	3/4 2-7
3/4 3-16	3/4 3-16
3/4 3-17	3/4 3-17
3/4 3-18	3/4 3-18
3/4 3-19	---
3/4 3-20	---
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2
3/4 7-3	3/4 7-3
B 3/4 7-1	B 3/4 7-1

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
3. Logarithmic Power Level - High (1)	≤ 0.75%	≤ 0.819%
4. Pressurizer Pressure - High	≤ 2362 psia	≤ 2370.887 psia
5. Pressurizer Pressure - Low	≥ 1675 psia (2)	≥ 1643.9 psia (2)
6. Containment Pressure - High	≤ 18.3 psia	≤ 18.490 psia
7. Steam Generator Pressure - Low	≥ 751 psia (3)	≥ 738.6 psia (3)
8. Steam Generator Level - Low	≥ 22.2% (4)	≥ 21.5% (4)

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 ≤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. 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 ≤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 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 ≤200 psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to 120.4×10^6 lbm/hr.

APPLICABILITY: MODE 1

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit at least once per 12 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤18.3 psia	≤18.490 psia
c. Pressurizer Pressure - Low	≥ 1675 psia	≥ 1643.9 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons	Not Applicable	Not Applicable
b. Containment Pressure - High-High	≤23.3 psia	≤23.490 psia
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual (Trip Buttons	Not Applicable	Not Applicable
b. Containment Pressure - High	≤18.3 psia	≤18.490 psia

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
4. MAIN STEAM AND FEEDWATER ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 751 psia (2)	≥ 738.6 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	≥ 1675 psia	≥ 1643.9 psia
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	(4)	2300 ± 699 volts with a 0.64 ± 0.34 second time delay
b. 460 volt Emergency Bus Undervoltage	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 SETPOINT	ALLOWABLE VALUES
8. EMERGENCY FEEDWATER (EFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator (A&B) Level - Low	≥ 22.2% (3)	≥ 21.5% (3)
c. Steam Generator ΔP-High (SG-A > SG-B)	≤90 psi	≤99.344
d. Steam Generator ΔP-High (SG-B > SG-A)	≤90 psi	≤99.344
e Steam Generator (A&B) Pressure - Low	≥ 751 psia (2)	≥ 738.6 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 before pressurizer pressure exceeds 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 narrow range level instrument nozzles.

(4) The trip value for this function is listed in the surveillance test procedures. The trip value will ensure that adequate protection is provided when all the applicable calibration tolerances, channel uncertainties, and time delays are taken into account.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-5.

APPLICABILITY: MODES 1, 2 and 3*

ACTION:

MODES 1 and 2

With one or more main steam line code safety valves inoperable, operation in MODES 1 and 2 may proceed provided that within 4 hours, power is reduced to less than or equal to the applicable percent of RATED THERMAL POWER as listed in Table 3.7-1 and within 12 hours, the Linear Power Level-High trip setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

MODE 3

With one or more main steam line code safety valves inoperable, operation in MODE 3 may proceed provided that at least 2 main steam line code safety valves are OPERABLE on each steam generator; otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

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

*Except that during hydrostatic testing in Mode 3, eight of the main steam line code safety valves may be gagged and two (one on each header) may be reset for the duration of the test to allow the required pressure for the test to be attained. The Reactor Trip Breakers shall be open for the duration of the test.

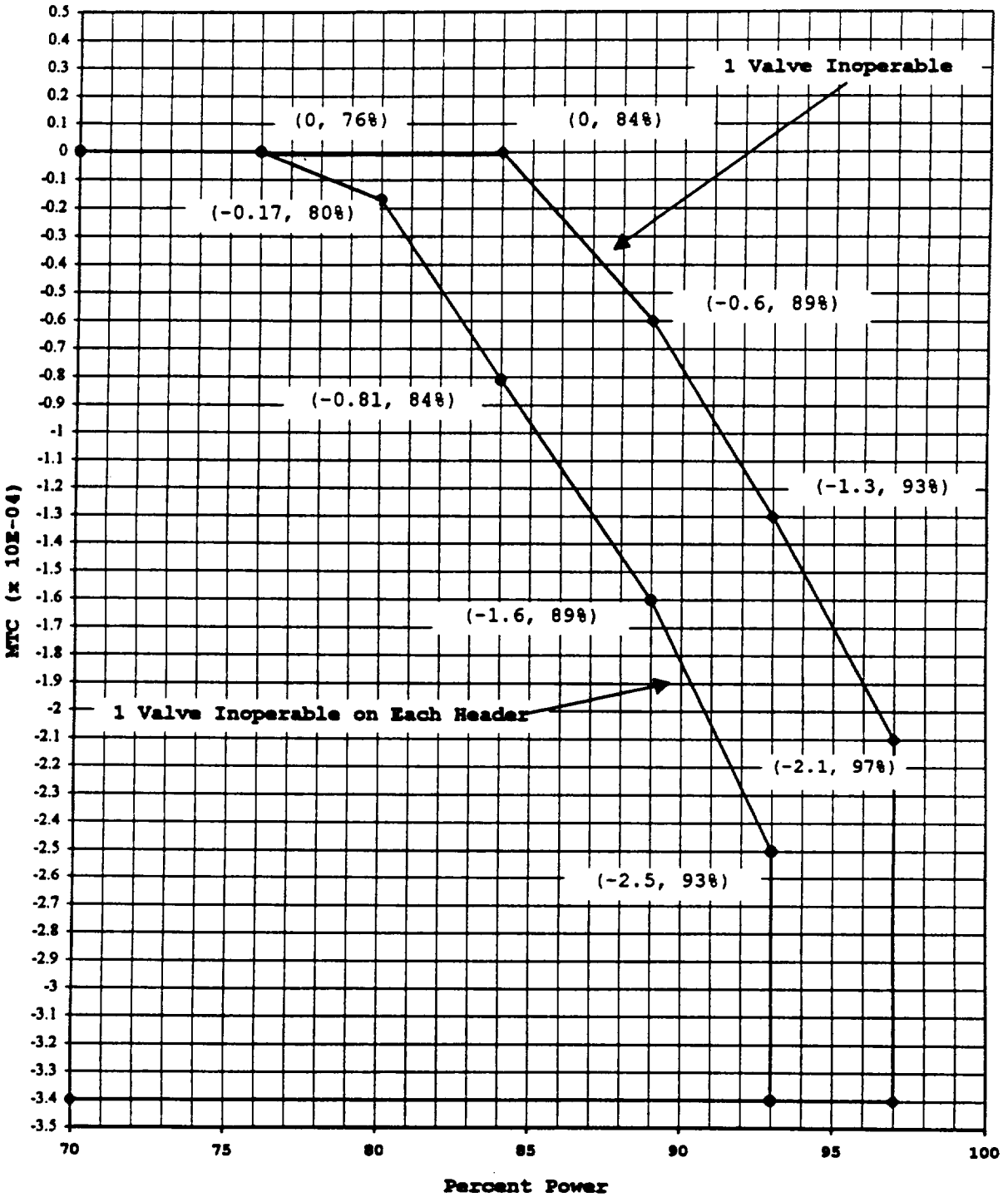
TABLE 3.7-1

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

Number of Inoperable Safety Valves	Maximum Allowable Linear Power Level And High Trip Setpoint (Percent of RATED THERMAL POWER)
1 Valve Inoperable	84% (except as allowed by Figure 3.7-1)
1 Valve Inoperable on Each Header	76% (except as allowed by Figure 3.7-1)
Maximum of 2 Valves Inoperable on Each Header	43.0
Maximum of 3 Valves Inoperable on Each Header	25.0

FIGURE 3.7-1

MTC Versus Maximum High Linear Power Level And Trip Setpoint



3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

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 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 capacity exceeds the 102% RATED THERMAL POWER (100% + 2% for instrument error) steam flow with steam pressure at 110% of the secondary system design pressure. 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 setpoint reductions are derived by an analysis of a loss of condenser vacuum event initiated at the reduced power levels listed in Table 3.7-1 that shows peak steam generator pressures are maintained below 110% of design pressure.

To provide power level limits more amenable to MSSV testing, the LOCV analysis also determines the combination of allowable initial power levels and moderator temperature coefficients (MTC) that yield acceptable results for the single most limiting valve and one bank of valves inoperable. These power level/MTC combinations are the basis of Figure 3.7-1.

The 4-hour completion time for required Action (a) is a reasonable time period to reduce power level and is based on the low probability of an event occurring during this period that would require activation of the MSSVs. An additional 8 hours is allowed in Action (a) to reduce the setpoints in recognition of the difficulty of resetting all channels of this trip function within a period of 4 hours. The completion time of 12 hours for Action (a) is based on operating experience in resetting all channels of a protective function and on the low probability of the occurrence of a transient that would result in steam generator overpressure during this period.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 222 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 November 29, 1999, as supplemented by letters dated January 26, May 17 (2 letters), May 31, and August 4, 2000, Entergy Operations, Inc. (Entergy or the licensee), submitted a request for changes to the Arkansas Nuclear One (ANO), Unit 2 (ANO-2), License, Technical Specifications (TSs), and Bases to maintain consistency with the transient and accident analyses which evaluated the impact of the replacement steam generators (SGs) that are being used for Cycle 15 operation. The License change would incorporate a new methodology employed in calculating radiological doses for some non-loss-of-coolant accident (LOCA) events, to resolve an unreviewed safety question that was identified by the licensee. TS changes would be made to the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) low pressurizer pressure setpoints, the RPS and ESFAS low SG pressure setpoints, the RPS and ESFAS low SG level setpoints, the reactor coolant flow rate limit, and the high linear power trip setpoints with inoperable main steam safety valves (MSSVs).

The amendment would also make changes to the TSs (and the Bases would also be changed to correspond to the amendment) that are not directly related to the replacement SGs. These changes revise the allowed outage time of the MSSVs in Modes 1 and 2 to allow up to 12 hours to reduce the high linear power level-high trip setpoint when one or more MSSVs are inoperable, and revise the action statement in Mode 3 to maintain at least two MSSVs operable on each SG.

The January 26, May 17 (2 letters), May 31, and August 4, 2000, letters provided clarifying information that did not change the scope of the original application and the initial proposed no significant hazards consideration determination.

2.0 TRANSIENT AND ACCIDENT ANALYSIS

2.1 Background

The licensee has proposed changes to TS Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits," TS Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values," TS 3.2.5, "RCS [Reactor Coolant System] Flow Rate," and

TS 3.7.1.1, "Turbine Cycle Safety Valves," with their associated Bases. Most of the proposed changes are required to maintain consistency with the transient and accident analyses which evaluated the impact of the replacement SGs (RSGs) that are being installed at ANO-2. However, the proposed changes to TS 3.7.1.1 are not directly related to the RSGs.

The proposed TSs are applicable to Cycle 15 operation with the RSGs at the current rated power of 2815 megawatts thermal (MWt).

2.2 Evaluation

The RSGs are a Westinghouse design, whereas the original SGs (OSGs) were designed by Combustion Engineering (CE). The new design has provided several enhancements to operational performances and has resulted in several changes compared to the OSGs, including increased number of SG tubes and increased primary and secondary side volumes. These differences have required evaluation and reanalysis of related design basis transient and accidents and subsequently resulted in the need for the revision of several TS limits.

The licensee's proposed changes to TS Table 2.2-1 and 3.3-4 will decrease the low pressurizer pressure setpoint from ≥1717.4 psia to ≥1675 psia, increase the low SG pressure setpoint from ≥712 psia to ≥751 psia, and decrease the low SG level from 23 percent to 22.2 percent. The proposed changes to TS 3.2.5, will increase RCS flow rate from 108.4×10^6 lbm/hr to 120.4×10^6 lbm/hr, which is the original value associated with the OSGs prior to tube plugging.

The licensee has performed an evaluation of the Safety Analysis Report (SAR) Chapter 15 safety analyses to assure that the current safety analyses are valid with the RSGs installed and the proposed TSs. For events in which the effects of the TS and RSG changes necessitated new analyses, reanalyses were performed to determine the impact of operation with the RSGs and the proposed TS restrictions. The list of events that have been reanalyzed include the large-break (LB) LOCA (LBLOCA) and small-break (SB) LOCA (SBLOCA), main steam line break (MSLB), feedwater line break (FWLB), loss of external load and/or turbine trip, loss of normal feedwater flow, excess heat removal due to secondary system malfunction, and instantaneous closure of a main steam isolation valve. In its reanalyses of non-LOCA events, the licensee has used approved computer codes and methodology essentially consistent with its licensing basis. There are some changes in input assumptions including reactor protection and engineered safety feature response times and the methods to determine a limiting break size for a design basis FWLB event. The staff has evaluated these modified input assumptions and finds that they are conservative and acceptable. The staff has concluded that the results of the licensee's reanalyses of non-LOCA events meet the acceptance criteria for each event.

In its original November 29, 1999, submittal and in a supplementary letter dated May 17, 2000 (2CAN050005), the licensee described the ANO-2 LOCA analyses to support the proposed TS changes.

The SBLOCA analyses reference the Asea Brown Broveri-Combustion Engineering (ABB-CE) SBLOCA evaluation model described in the topical report CENPD-137, Supplement 2, P-A. CENPD-137, Supplement 2, P-A, has been generically approved as applicable to CE reactor designs. ABB-CE identifies to its licensees those analysis input parameters to which LOCA analysis results are sensitive. The supplemental letter showed that the CENPD-137, Supplement 2, P-A methodology applies to ANO-2 by describing ongoing processes used by

Entergy and CE (its fuel vendor) to determine values for analysis input parameters which assure that the input values bound the actual as-operated plant values for peak cladding temperature (PCT)-sensitive parameters.

The November 29, 1999, submittal indicates that the LBLOCA analyses was performed with the generically approved June 1985 ABB-CE LBLOCA methodology. The May 17, 2000 (2CAN050005), supplemental letter identifies that the ongoing Entergy/CE input value determination processes are also used to specify LBLOCA input values to assure that the input values bound the actual as-operated plant values for PCT-sensitive parameters.

Based on the generic acceptability and applicability of the SBLOCA and LBLOCA models, and on the licensee's description of plant-specific processes it will use to assure that PCT-sensitive parameter values input to the SBLOCA and LBLOCA models bound the actual as-operated plant values for ANO-2, the staff concludes that the SBLOCA and LBLOCA methodologies discussed above apply to ANO-2. Therefore, these methodologies are acceptable for use in performing ANO-2 SBLOCA and LBLOCA analyses, and they continue to be acceptable for reference in the ANO-2 TSs and the core operating limits report (COLR).

The November 29, 1999, submittal includes analysis assumptions and LBLOCA and SBLOCA results. During its review, the staff referred to these as examples; but the examples were not within the formal scope of the review, because including analytical methodologies, as supported by the data input processes discussed above in the TSs and the COLR, assure that the analyses will be properly performed and used in governing plant operation.

The licensee's proposed modification of TS 3.7.1.1, "Turbine Cycle Safety Valves," will revise Action (a) to allow up to 12 hours to reduce the High Linear Power Level in accordance with Table 3.7.1 when one or more MSSVs are inoperable. In addition, this action has been revised to be applicable in Modes 1 and 2 only. The shutdown requirement of this action has also been revised to require a shutdown to Mode 4 within an additional 12 hours. The above revisions are consistent with the current standard TSs in NUREG-1432. Action (b) of TS 3.7.1.1 is revised to be applicable to Mode 3 only since operation with less than two reactor loops in Modes 1 and 2 is not allowed at ANO-2. Also, a new TS Figure 3.7-1, "MTC [Moderator Temperature Coefficient] Versus Maximum High Linear Power Level and Trip Setpoint," has replaced the deleted Table 3.7-2 to provide a comparison of the percent of rated power versus the MTC as a method of determining the value for the linear power level during periods when a MSSV is inoperable. The application of this new figure for conditions where not more than one MSSV is inoperable on each steam header will allow testing of the MSSVs without a significant decrease in power level. The proposed changes also reduce the allowable linear power level during conditions of two or more MSSVs being inoperable on each steam header. The results of the licensee's safety analyses confirms that the above proposed changes of allowable high linear power level are conservative.

2.3 Summary

The staff has reviewed the licensee-proposed changes discussed above. The staff finds that the licensee's proposed TSs will not affect safe operation of ANO-2 and, therefore, finds the proposed TSs with their associated Bases acceptable.

3.0 SETPOINT ANALYSIS

3.1 Background

The licensee stated that the existing SGs were procured from CE, while the replacement SGs are of Westinghouse design. The difference between the two designs required reanalysis of related design-basis accidents and subsequently resulted in the need for revision of several TS limits. Set forth below are the proposed changes related to RPS- and ESFAS-related instrumentation setpoint and allowable values.

3.2 Evaluation

3.2.1 TS Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits."

1. For Functional Unit 5, Pressurizer Pressure - Low, change the Trip Setpoint from ≥1717.4 psia to ≥1675 psia and the Allowable Value from ≥1686.3 psia to ≥1643.9 psia.

2. For Functional Unit 7, Steam Generator Pressure - Low, change the Trip Setpoint from ≥712 psia to ≥751 psia and the Allowable Value from ≥ 699.6 psia to ≥738.6 psia.

3. For Functional Unit 8, Steam Generator Level - Low, change the Trip Setpoint from ≥23% to ≥22.2% and the Allowable Value from ≥22.111% to ≥21.5%.

3.2.2 TS Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values."

1. For Functional Unit 1c, Pressurizer Pressure - Low, change the Trip Setpoint from ≥1717.4 psia to ≥1675 psia and the Allowable Value from ≥1686.3 psia to ≥1643.9 psia.

2. For Functional Unit 4b, Steam Generator Pressure - Low, change the Trip Setpoint from ≥712 psia to ≥751 psia and the Allowable Value from ≥699.6 psia to ≥738.6 psia.

3. For Functional Unit 5c, Pressurizer Pressure - Low, change the Trip Setpoint from ≥1717.4 psia to ≥1675 psia and the Allowable Value from ≥1686.3 psia to ≥1643.9 psia.

4. For Functional Unit 8b, Steam Generator (A&B) Level - Low, change the Trip Setpoint from ≥23% to ≥22.2% and the Allowable Value from ≥22.111% to ≥21.5%. Also, Note (3), which is applicable to this setpoint, is clarified as a narrow range instrument.

5. For Functional Unit 8e, Steam Generator (A&B) Pressure - Low, change the Trip Setpoint from ≥712 psia to ≥751 psia and the Allowable Value from ≥699.6 psia to ≥738.6 psia.

Evaluation: All of the above-described changes are based on the revised safety analyses performed for implementation of the new SGs replacing the existing degraded SGs. In its submittal, the licensee stated that the revised setpoints and allowable values were determined using the in-house setpoint calculation methodology. In a supplemental letter dated May 31, 2000, the licensee stated that: "The setpoint methodology utilized to calculate changes to the reactor protective system and engineered safety features actuation system related instrumentation setpoints (reference the November 29, 1999, submittal) is the same setpoint methodology reviewed and approved by the NRC in License Amendments 137 and 138 regarding the containment building pressure and pressurizer pressure trip setpoints, respectively." Since the licensee has used the staff approved calculation methodology, the staff finds the proposed changes acceptable.

3.2.3 TS Table 2.2-1, "Reactor Protective Instrumentation Trip Setpoint Limits," Functional Unit 2, Linear Power Level - High.

The licensee proposes to delete items 2b, 2c, and 2d completely, including a symbol "*" shown in columns for Trip Setpoint and Allowable Values, along with the footnote at the bottom of page 2-5, which reads as follows:

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

Evaluation: In its submittal, the licensee stated that these items are being deleted as part of a page cleanup effort. No plans currently exist for ANO-2 to address startup and/or power operation with fewer than four reactor coolant pumps operating. Therefore, the aforementioned items have been deleted. This change is acceptable to the staff.

3.2.4 TS Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values"

The licensee proposes to modify the footer on page 3/4 3-18, so that the next page is now referred to as "3/4 3-21". Also, pages 3/4 3-19 and 3/4 3-20, which in the current TS are intentionally left blank, are being removed from the TS.

Evaluation: The proposed change is editorial in nature and is, therefore, acceptable to the staff.

3.3 Summary

On the basis of this evaluation, the staff concluded that the licensee has revised setpoint and allowable values based on the previously staff-acceptable calculation methodology. Therefore, the staff finds the above mentioned proposed changes acceptable.

4.0 RADIOLOGICAL DOSE ASSESSMENT

4.1 Background

The RSGs are of a different design than the OSGs. The licensee proposed several TS changes intended to maintain consistency with the transient and accident analyses performed to evaluate the impact of the RSG installation. The installation of the RSGs results in the following changes that were accommodated in the re-analyses performed by the licensee:

- The RSGs are larger than the OSGs, resulting in a larger secondary side mass.
- The larger RSGs contain more tubes that the OSGs, resulting in increased RCS mass.
- The larger RSGs have increased RCS flow and primary-to-secondary heat transfer area, resulting in higher SG operating pressures.
- The RSG design incorporates an internal steam flow-limiting nozzle, resulting in a reduction in the effective cross-sectional area of the steam line.

In addition to these changes in analyses parameters, the licensee incorporated the following analysis changes:

- A core power of 3087 MWt (3026 MWt with 1.02 factor) in the radiological analyses of the seized rotor accident and the MSLB. This is conservatively greater than the current licensed power and is consistent with a planned future power uprate.
- Several changes were made to the methodologies used in assessing the radiological consequences of the affected accidents.

4.2 Evaluation

In evaluating the consequences of the replacement of the SGs, the licensee performed an engineering review of the various accident analyses described in the ANO-2 SAR to determine the effect(s) of the SG replacements. Where effects were identified, the licensee performed further evaluations and, where needed, re-analyses. This safety evaluation (SE) section addresses the radiological consequences of these effects and whether these consequences continue to meet dose guidelines and criteria. The licensee identified five affected radiological analyses:

1. SG Tube Rupture (SGTR)
2. Loss of AC Power Auxiliaries (LOAC)
3. Main Coolant Pump Seized (locked) Rotor (LRA)
4. FWLB
5. MSLB

The staff reviewed the results of the licensee's categorization of analysis impacts and, based on its experience with similar applications, finds this listing adequately encompassing. The staff did note, however, that the licensee did not address the potential consequences on post-accident control room operator doses in the docketed material. This issue was discussed with the licensee during teleconferences on April 3, 2000, and April 20, 2000. The licensee responded by letter dated May 17, 2000 (2CAN050006). The staff did not find that response

acceptable. By letter dated August 4, 2000, the licensee provided additional information. This issue will be further discussed later in the SE.

4.2.1 SGTR

An SGTR accident involves the rupture of a SG tube, resulting in the transfer of reactor coolant to the secondary side. A loss-of-offsite power is conservatively assumed. Because of the loss of AC power, radioactivity is released from the SGs via steam relief valves and atmospheric dump valves to the environment.

The licensee evaluated the potential impact on the prior SGTR analysis and concluded that no re-analysis was necessary. The licensee stated that the smaller U-tube diameter, higher secondary side pressure, and more restrictive tube flow resistances of the RSGs would result in a lower primary-to-secondary mass transfer than previously estimated. The licensee concluded that the prior analyses with the higher postulated mass transfer would bound the consequences of a SGTR occurring in an RSG, and that no further re-analysis was necessary. The staff finds this conclusion acceptable.

4.2.2 LOAC

A LOAC accident results from an event in which offsite AC power is lost. From a radiological standpoint, this event results in a protracted plant cooldown effected by the dumping of steam to the environment via SG atmospheric dump valves. It is conservatively assumed that reactor coolant and SG leakage and specific activity are at their maximum TS values.

The licensee evaluated the potential impact on the prior LOAC analysis and concluded that no re-analysis was necessary. The licensee stated that the radiological consequences of this event are bounded by the analyses for the MSLB and FWLB accidents and no further re-analysis was necessary. The staff finds this conclusion acceptable.

4.2.3 LRA

An LRA results from a mechanical defect that causes the impeller of a main coolant pump in a reactor coolant loop to suddenly stop rotating. This substantially reduces reactor coolant flow in the affected loop resulting in a sharp reduction in reactor coolant pressure in localized regions of the core. Fuel damage is postulated.

The licensee re-analyzed the LRA using the increased reactor coolant and SG liquid masses, the increased reactor power, and the changes in dose calculational methodology. Since a specific fuel damage estimate was not available for the Cycle 15 fuel load, the licensee performed a reiterative analysis to determine the amount of fuel damage that could occur with doses still remaining within dose acceptance criteria. From this evaluation, the licensee postulated a fuel damage of 14 percent. In their letter of May 17, 2000, the licensee stated that the preliminary results of Cycle 15 thermo-dynamic analyses indicate that the estimated fuel damage would be less than three percent. The licensee stated that the results of their analyses indicated that offsite doses would remain within a small fraction (10 percent) of the 10 CFR Part 100 accident dose guidelines for estimated fuel damage up to 14 percent. In their letter dated August 4, 2000, Entergy stated that the re-analysis of the control room doses from a LRA

would be within the dose criteria of 10 CFR Part 50, Appendix A, General Design Criterion (GDC)-19. See the discussion in Section 4.2.6 below.

The staff reviewed the description of the LRA accident analysis and performed an independent analysis to confirm the licensee's conclusions. The staff finds the licensee's conclusion that the dose consequences calculated for this accident will be within those permitted by the accepted dose guidelines, with fuel damage less than 14 percent, acceptable.

4.2.4 FWLB

A FWLB results from a break in a feedwater line leading to one of the SGs. This break causes a depressurization of the affected SG and a blowdown of its liquid contents through the break. Radioactivity is released to the environment via the break and from steam dumping associated with the plant cooldown.

The licensee re-analyzed the FWLB using the increased reactor coolant and SG liquid masses, and the changes in dose calculational methodology. The licensee did not use the increased reactor power in the radiological consequence analyses associated with this event. The licensee stated that the limiting break would occur inside containment since check valves located on the feedwater line would prevent the SGs from blowing down outside. The licensee did not credit holdup in the containment in assessing the releases. The licensee did not assume an iodine spike, as this assumption is not consistent with its licensing basis. The licensee stated that the results of their analyses indicated that offsite doses would remain within a small fraction (10 percent) of the 10 CFR Part 100 accident dose guidelines. In their letter dated August 4, 2000, the licensee stated that the re-analysis of the control room doses from a FWLB would be bounded by the doses postulated for the MSLB. See the discussion in Section 4.2.6 below.

The staff reviewed the description of the FWLB accident analysis and performed an independent analysis to confirm the licensee's conclusions. The staff concludes that the dose consequences calculated for this accident will be within those permitted by the accepted dose guidelines.

4.2.5 MSLB

An MSLB results from a break in a main steam line leading from one of the SGs. This break causes a depressurization of the affected SG and a blowdown of its contents through the break. Radioactivity is released to the environment via the break and from steam dumping associated with the plant cooldown.

The licensee re-analyzed the MSLB using the increased reactor coolant and SG liquid masses, the increased reactor power, and the changes in dose calculational methodology. For the MSLB break, the licensee considered pre-incident and accident-initiated iodine spikes and assumed that the break would occur outside of containment but upstream of the isolation valve and that the affected SG would boil dry. Plant cooldown would be achieved by dumping steam from the unaffected SG. The licensee stated that the results of their analyses indicated that offsite doses would remain within a small fraction (10 percent) of the 10 CFR Part 100 accident dose guidelines for the accident-initiated iodine spike case and would remain within 10 CFR Part 100 accident dose guidelines for the pre-incident iodine spike case. In their letter dated

August 4, 2000, the licensee stated that the re-analysis of the control room doses from an MSLB would be within the dose criteria of 10 CFR Part 50, Appendix A, GDC-19. See the discussion in Section 4.2.6 below.

In early 1999, another power reactor licensee submitted a licensee event report (LER) regarding non-conservatisms in the assessment of the iodine spike appearance rate that is used in evaluating the accident-initiated iodine spike case. This LER identified that the original vendor assessments assumed a letdown system flow rate less than the maximum possible system flow rate. The staff identified that prior analyses for ANO-2 assumed a flow rate of 40 gpm, about one-third of the maximum system flow of 128 gpm. At ANO-2, the letdown flow rate is a function of the number of charging pumps operating. The staff requested that the licensee address the applicability of the LER issues to ANO-2. In its response of August 4, 2000, the licensee stated that while plant procedures do not prohibit operation of additional charging pumps, operation with more than one charging pump during steady-state operations is rare. Since operation at the higher flow rate is an infrequent situation, the staff finds acceptable the licensee's position that an iodine appearance rate assessment based on the normal letdown flow rate is appropriate.

The staff reviewed the description of the MSLB accident analysis and performed an independent analysis to confirm the licensee's conclusions. The staff finds that the dose consequences calculated for this accident will be within those permitted by the accepted dose guidelines.

4.2.6 Control Room Operator Doses

The licensee did not address the impact of the SG replacement on the post-accident doses to the control room operators in its original submittal. In conference calls held on April 3, 2000, and April 20, 2000, the staff requested the licensee to: (1) address the impact of the proposed changes on control room habitability, and (2) provide an explanation supporting the conclusion that the control room unfiltered in-leakage is limited to only 10 cfm as stated in the SAR.

In their letter dated May 17, 2000 (2CAN050006), the licensee responded to both of these information requests. With regard to the first issue, the licensee provided a discussion on their evaluation that establishes the LOCA analysis as the limiting accident (the radiological consequences of a LOCA are not affected by the SG replacement). The licensee addressed accident-specific differences in atmospheric dispersion, radioactivity release rate, and control room isolation actuation.

The licensee's response to the issue of control room infiltration did not provide a basis for understanding why 10 cfm was an appropriate assumption. The ANO in-leakage assumption of 10 cfm is questionable in light of the industry experience with recent in-leakage testing. In-leakage testing at 20 percent of the current operating plants has shown measured infiltration rates exceeding the values assumed in the design basis analyses. In its response, the licensee took the position that LOCA consequences were bounding and that they were pursuing a generic industry resolution to the in-leakage issue. Since the licensee did not provide sufficient information for the staff to find that the LOCA consequences were bounding for the MSLB, FWLB, and LRA accident analyses, and since the staff can not make a current finding of acceptability on the basis of future compliance with a voluntary industry initiative, this response was not accepted.

By letter dated August 4, 2000, the licensee submitted a description of the revised control room dose calculations for the MSLB, FWLB, and LRA accident analyses. These analyses assumed an unfiltered in-leakage flow rate of 5000 cfm. Although not based on the results of in-leakage testing, the licensee stated their belief that the 5000 cfm value was a bounding number. This conclusion was based on the relative volume of the ANO control room in relation to the assumed in-leakage. A control room air exchange of once every eight minutes is implied by this assumption. The licensee noted that the American Society of Heating Refrigeration and Air Conditioning Engineers guidance states that typical control room envelopes have air exchanges on the order of once per hour, or about 667 cfm for ANO. The licensee reported the limiting dose consequence (MSLB) to be 21.4 rem with the iodine protection factor (IPF) of 1.38. An IPF of 1.0 (i.e., no credit for control room envelope) would have resulted in a dose of 29.6 rem which is less than the 30 rem thyroid acceptance criteria for control room habitability.

In their response, the licensee noted that they (1) were actively participating in the industry initiative, (2) have initiated a condition report to track and resolve the generic issue, (3) will be giving further consideration to control room infiltration as part of a power uprate submittal scheduled to be presented this year, and (4) have implemented corrective and preventative programs for control room integrity components. Based on its re-analyses with the assumed 5000 cfm in-leakage and the other information, the licensee believes that the setpoint changes associated with the SG replacement may be approved without imposing significant risk to the public or control room personnel.

Based on its review of the additional information docketed by the licensee, the staff has determined that there is reasonable assurance that the ANO control room would be habitable during a MSLB, FWLB, or LRA accident and that this amendment may be approved before the resolution of the generic issues. The staff bases this determination on the results of the licensee's analyses that assumed an infiltration of 5000 cfm, and the expectation that 5000 cfm is bounding. The approval of this amendment does not exempt the licensee from regulatory actions that may be imposed in the future as this generic issue is resolved.

4.2.7 Revision to Control Room Atmospheric Dispersion Values

In re-analyzing the control room doses, the licensee used atmospheric dispersion values, χ/Q, that were generated as part of this effort. The licensee used the Nuclear Regulatory Commission (NRC)-sponsored computer code ARCON96 to generate these data. These new values were determined based on five years of site-specific meteorological data collected between January 1995 and December 1999 from the site meteorological tower. All releases were considered to be ground level. An attachment to the August 4, 2000, letter provides a description and derivation for each of the computer code inputs. The χ/Q values (sec/m^3) determined are:

	Atmospheric Dump Valve		MSSV		MSLB	
	Intake 1	Intake 2	Intake 1	Intake 2	Intake 1	Intake 2
0 to 2 hrs	6.31E-4	4.78E-4	8.05E-4	5.91E-4	5.48E-4	4.22E-4
2 to 8 hrs	3.65E-4	2.75E-4	4.64E-4	3.37E-4	3.23E-4	2.51E-4

The staff reviewed the use of the ARCON96 computer code and the code input information provided by the licensee and found the code use and code inputs to be consistent with current

staff positions on the use of ARCON96. The staff performed a qualitative review of the code results and deemed them to be reasonable. The staff finds the revised χ/Q values acceptable.

4.3 Summary

Based on the information provided by the licensee related to the proposed changes in setpoints associated with replacement of the SGs, the staff finds reasonable assurance that the radiological consequences of anticipated accidents at ANO-2 will continue to be less than the dose guidelines of 10 CFR Part 100 and the criteria of 10 CFR Part 50, Appendix A, GDC-19 and Section 6.4, "Control Room Habitability System," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."

5.0 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.

6.0 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. 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 (65 FR 6405, dated February 9, 2000). 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.

7.0 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.

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