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

November 13, 2000

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: TECHNICAL SPECIFICATION CHANGES AND UNREVIEWED SAFETY QUESTION RESOLUTION RELATED TO CONTAINMENT BUILDING DESIGN PRESSURE INCREASE TO 59 PSIG (TAC NO. MA7044)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No. 225 to Facility Operating License No. NPF-6 for Arkansas Nuclear One, Unit No. 2 (ANO-2). This amendment authorizes changes to the Safety Analysis Report (SAR) and changes the Technical Specifications (TSs) in response to your application dated November 3, 1999, as supplemented by letters dated April 4, June 9, June 29, August 2, and August 16, 2000.

The amendment authorizes revision of the SAR to increase the containment structural design pressure from 54 psig to 59 psig, revises TS Table 3.3-3 to add a containment spray actuation signal on high-high containment building pressure to terminate main feedwater and main steam flow from the unaffected steam generator, revises TS 3.6.1.4 and Figure 3.6-1 to change the allowable containment initial conditions to be consistent with analysis assumptions, and revises TS 6.15 to increase the calculated peak accident pressure in the containment leakage rate testing program from 54 psig to 58 psig. Related changes to the Bases were also made.

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. 225 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

November 13, 2000

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Entergy Operations, Inc.
1448 S. R. 333
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Thomas W. Alexion

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. 225 to NPF-6
2. Safety Evaluation

cc w/encls: See next page

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. 225
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 3, 1999, as supplemented by letters dated April 4, June 9, June 29, August 2, and August 16, 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. 225 , the Facility Operating License No. NPF-6 is amended to authorize revision of the Safety Analysis Report (SAR) to increase the containment structural design pressure from 54 psig to 59 psig as set forth in the application for amendment by Entergy Operations, Inc., dated November 3, 1999, as supplemented by letters dated April 4, June 9, June 29, August 2, and August 16, 2000, and evaluated in the staff's safety evaluation enclosed with this amendment. Entergy Operations, Inc., shall incorporate the revision into the next SAR update in accordance with the schedule in 10 CFR 50.71(e).

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. 225 , 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 the commencement of heatup from refueling outage 2R14.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Gramm

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: November 13, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 225

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
3/4 3-11	3/4 3-11
3/4 6-6	3/4 6-6
3/4 6-7	3/4 6-7
6-26	6-26
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	9
b. Containment Pressure - High	4	2	3	1, 2, 3	10,11
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	10,11
d. ESFAS Logic					
1. Matrix Logic	6	1	3	1, 2, 3	12
2. Initiation Logic	4	2	4	1, 2, 3, 4	9
e. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13
2. CONTAINMENT SPRAY, MAIN STEAM, AND MAIN FEEDWATER ISOLATION (CSAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	9
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	10,11
c. ESFAS Logic					
1. Matrix Logic	6	1	3	1, 2, 3	12
2. Initiation Logic	4	2	4	1, 2, 3, 4	9
d. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	13

INTERNAL PRESSURE AND AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 The combination of containment internal pressure and average air temperature shall be maintained within the region of acceptable operation shown on Figure 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

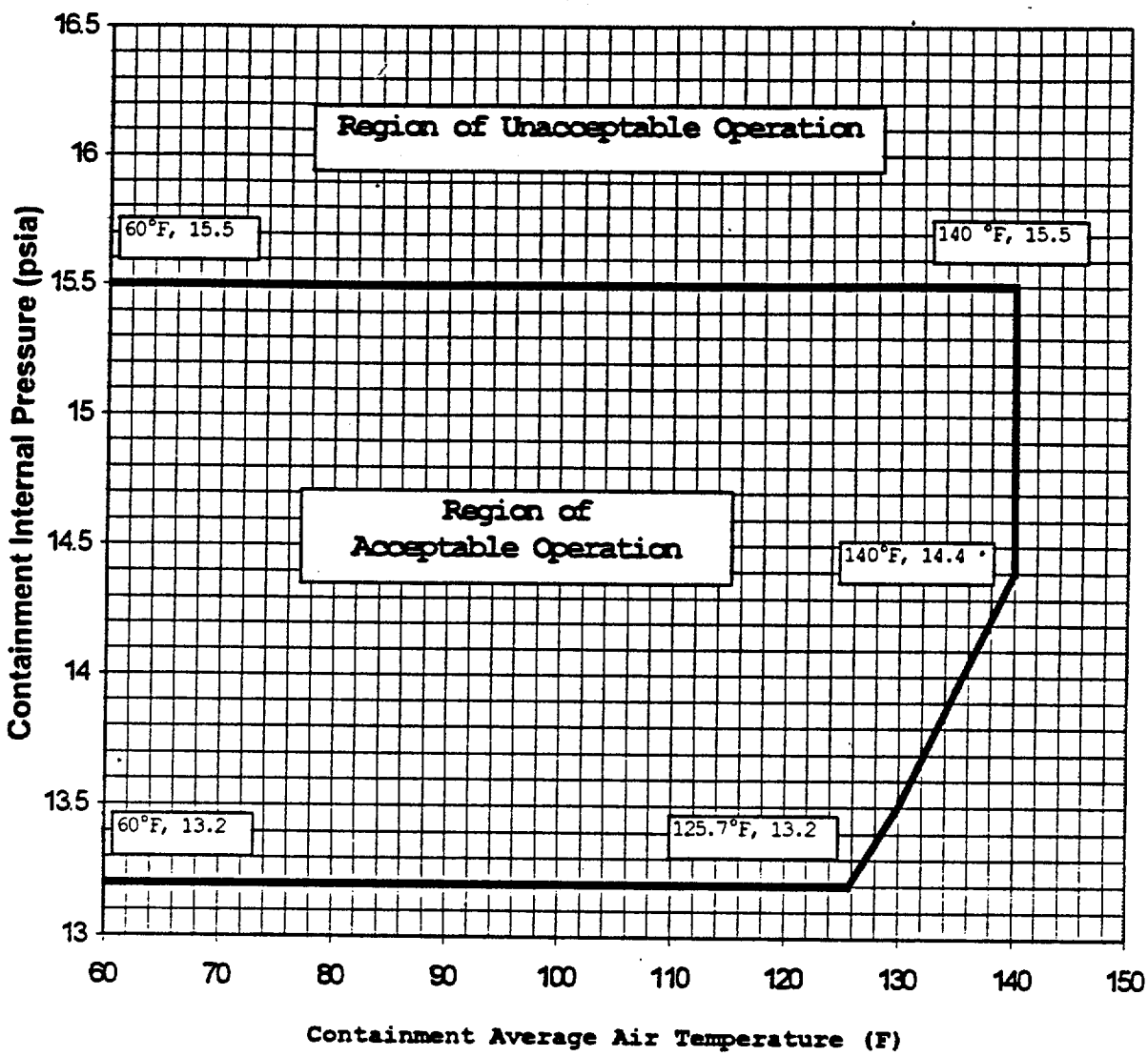
With the point defined by the combination of containment internal pressure and average air temperature outside the region of acceptable operation shown on Figure 3.6-1, restore the combination of containment internal pressure and average air temperature to within the above limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure and average air temperature shall be determined to be within the limits at least once per 12 hours. The containment average air temperature shall be the temperature of the air in the containment HVAC common return air duct upstream of the fan/cooler units.

FIGURE 3.6-1

CONTAINMENT INTERNAL PRESSURE VS. AVERAGE AIR TEMPERATURE



NOTE: Instrument Error is not Included on Curve

6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 58 psig.

The maximum allowable containment leakage rate, L_a, shall be 0.1% of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. Leakage rate for each door is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak design basis loss of coolant accident pressure, P_a, of 58 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ during the performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Option B of Appendix "J" of 10 CFR 50.

The containment will be periodically leakage tested in accordance with the Containment Leakage Rate Testing Program. These periodic testing requirements verify the containment leakage rate does not exceed the assumptions used in the safety analysis. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$.

3/4.6.1.3 CONTAINMENT AIR LOCKS

Each containment air lock forms part of the containment pressure boundary. As part of the containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event. For the purposes of this specification, the vertical end plates of the air lock barrel, on which the doors themselves are mounted, shall be considered part of the door.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

3/4.6.1.4 INTERNAL PRESSURE AND AIR TEMPERATURE

The limitations on containment internal pressure and average air temperature, assuming a worst case relative humidity value of 0 %, ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 5.0 psi, 2) the containment peak pressure does not exceed the design pressure of 59 psig during design basis conditions, 3) the ECCS analysis assumptions are maintained, and 4) the containment cooling fan motor qualifications are maintained.

The limitation on containment average air temperature ensures that the containment liner plate temperature does not exceed the design temperature of 300°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses. Figure 3.6-1 represents analysis limits and does not account for instrument error.

3/4.6.1.5 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum design pressure of 59 psig. The visual examination of tendons, anchorages and containment surfaces and the Type A leakage tests of the Unit 2 containment in conjunction with the required surveillance activities of the Unit 1 containment are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures", January 1976.

3/4.6.1.6 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 225 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 3, 1999, as supplemented by letters dated April 4, June 9, June 29, August 2, and August 16, 2000, Entergy Operations, Inc. (the licensee), submitted a request for changes to the Arkansas Nuclear One, Unit No. 2 (ANO-2), licensing basis and Technical Specifications (TSs). The requested changes would revise the licensing basis to increase the containment structural design pressure from 54 psig to 59 psig, revise TS Table 3.3-3 to add a containment spray actuation signal on high-high containment building pressure to terminate main feedwater and main steam flow from the unaffected steam generator (SG), revise TS 3.6.1.4 and Figure 3.6-1 to change the allowable containment initial conditions to be consistent with analysis assumptions, and revise TS 6.15 to increase the calculated peak accident pressure in the containment leakage rate testing program from 54 psig to 58 psig. Related changes to the Bases were also requested.

The April 4 and June 9, 2000, supplemental letters responded to questions from the staff. The June 29, 2000, supplemental letter corrected proposed changes to TS Figure 3.6-1 as a result of corrections to input assumptions used in the computer code to perform the containment analysis, and withdrew the proposed TS change to clarify the allowable containment leakage rate. The August 2, 2000, supplemental letter corrected an administrative error in the revised TS page provided in the June 29, 2000, supplemental letter that was intended to reflect the withdrawal of the clarification of the allowable leakage rate. The August 16, 2000, supplemental letter responded to questions from the staff and withdrew the proposed TS change to increase the allowable containment spray pump degradation. The April 4, June 9, June 29, August 2, and August 16, 2000, supplemental letters provided clarifying information and revisions to the proposed TSs that did not change the scope of the original Federal Register notice or the initial no significant hazards consideration determination.

2.0 BACKGROUND

The licensee intends to replace both ANO-2 SGs with SGs that have a larger primary and secondary inventory compared to the original SGs. They also have a greater heat transfer

area, higher secondary side operating pressure, a larger metal mass and a flow limiting device in the steam outlet nozzle. Enclosure 6 of the licensee's November 3, 1999, submittal provides a more detailed comparison between the original and the replacement SGs. In addition, the licensee intends to operate in the future at a power level 7.5% above the currently licensed power. Although not part of this license amendment, the licensee has chosen to perform the containment accident analyses at this higher power level. These analyses, which assume the initial pressure and temperature conditions of TS 3.6.1.4, Figure 3.6-1, predict a peak accident pressure of 57.6 psig (rounded by the licensee to 58 psig for P_a[1]) which is above the current containment design pressure of 54 psig. The licensee has therefore increased the design pressure of the containment building from 54 psig to 59 psig. The licensee considers this change in containment design pressure to be an unreviewed safety question (USQ) in compliance with 10 CFR 50.59(a)(2) since the licensee considers the change to be a reduction in the margin of safety as described in the bases of the TSs. The licensee states that this is within the inherent design margin of the containment.

Several TS changes were also proposed, as follows:

- TS 3/4.3.2, Table 3.3-3, "Engineered Safety Feature Actuation System Instrumentation," is revised to reflect the addition of a containment spray actuation signal (CSAS) on high-high containment building pressure (25.7 psig) to selected components to isolate main feedwater and terminate main steam flow from the unaffected SG.

- TS 3.6.1.4, "Internal Pressure and Air Temperature" including Figure 3.6-1, "Containment Internal Pressure vs. Average Air Temperature," is revised to extend the region of acceptable operation for the containment atmosphere. In addition, initial humidity of the containment atmosphere is eliminated as a parameter. (It is conservatively assumed to be zero in the containment accident analyses for peak pressure and temperature.) The licensee's June 29, 2000, supplemental letter revised this curve from that given in the licensee's November 3, 1999, application.

- TS 3/4.6.2, "Depressurization, Cooling and pH Control Systems," is revised in the licensee's November 3, 1999, application to increase the percentage degradation of the containment spray pump flow from 6.3% to 10%. In a supplemental letter dated August 16, 2000, the licensee withdrew the request for this proposed change.

- TS 6.15, "Containment Leakage Rate Testing Program," would be revised to reflect an increased calculated peak containment pressure, P_a, as a result of a loss-of-coolant accident (LOCA). In addition, in the licensee's November 3, 1999, submittal, L_a, the allowable containment leakage rate, which is defined in the TSs as 0.1 weight-% per day at P_a, was to be clarified to be to be equivalent

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[1] P_a is defined in Appendix J to 10 CFR Part 50 as the calculated peak internal pressure related to the design basis accident. It is the pressure at which containment leakage rate testing must be performed.

to 0.1 volume-% per day. By supplemental letter dated June 29, 2000, the licensee withdrew the clarification portion of the proposed change.

There are two postulated design basis accidents which can impact the containment integrity. These are the LOCA and the main steam line break (MSLB) accident. The licensee has re-analyzed both events. They are discussed separately below.

The licensee has re-analyzed the containment pressure and temperature response to the design basis LOCA based on the replacement SGs (RSGs) and other changes discussed in the licensee's November 3, 1999, application and June 29, 2000, supplement. The licensee performed the necessary calculations with Nuclear Regulatory Commission (NRC)-approved computer codes.

The results of the LOCA calculations are given in Table 6.2-8G of the licensee's June 29, 2000, supplement. The results show that the peak containment pressure and temperature following a LOCA remain below the containment design limits.

The licensee has re-analyzed the containment pressure and temperature response to the design basis MSLB accident based on the RSGs and the other changes discussed in the licensee's November 3, 1999, application and June 29, 2000, supplement. The licensee used NRC-approved computer codes to perform these calculations. The results of these analyses show that the peak pressure and temperature are below the new containment design pressure and temperature limits.

The licensee's June 29, 2000, supplement provided revised analyses of the LOCA and MSLB accident due to corrections made to two assumptions used in the COPATTA containment analysis computer code. The licensee reported that a correction was made to the surface area assumed for the containment heat sink and the time required to fill the containment spray headers, and thus the containment spray response time. The licensee stated that these changes would have increased the calculated peak containment pressure and temperature. The licensee therefore offset these changes by other changes to the input and methodology used to calculate containment pressure and temperature. These offsetting changes include additional containment heat sinks and changes to the initial conditions given in Figure 3.6-1 of the TSs. In addition, as discussed below, credit was taken in the MSLB accident analyses for 8% revaporization of condensate. This latter change is in accordance with the guidance of NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Appendix B. The results for the MSLB accident are given in Table 6.2-9C of the licensee's June 29, 2000, supplement. The results show that the peak pressures and temperatures following a MSLB accident are below the containment design limits.

3.0 EVALUATION

3.1 TS Changes and USQ

3.1.1 TS 3/4.3.2, Table 3.3-3

TS 3/4.3.2, Table 3.3-3, "Engineered Safety Feature Actuation System Instrumentation," is revised to reflect additional actuations on a CSAS. The CSAS is initiated on a high-high containment building pressure (25.7 psig). The additional actuations isolate main feedwater and terminate main steam flow from the unaffected SG in addition to actuating the containment spray system. The licensee has modeled these actions in the LOCA and MSLB accident calculations and has demonstrated that the containment design pressure and temperature are not exceeded.

In addition, the staff has reviewed the CSAS control circuits provided by the licensee by their supplemental letter dated April 4, 2000, and finds that they are in conformance with circuits used for other similar engineered safety features actuation system (ESFAS) circuits that the NRC reviewed in NUREG-0308, "Safety Evaluation Report, Arkansas Nuclear One, Unit 2," dated November 1977. The TS requirements in Table 3.3-3 for the modified Functional Unit 2 are the same as those for the original Functional Unit 2 (which included only containment spray) and other similar ESFAS functional units, and are therefore acceptable to the staff.

Based on the reasons discussed above, the staff finds this change to be acceptable.

3.1.2 TS 3.6.1.4 (including Figure 3.6-1)

TS 3.6.1.4, "Internal Pressure and Air Temperature," including Figure 3.6-1, "Containment Internal Pressure vs. Average Air Temperature," is revised to extend the region of acceptable operation for the containment atmosphere. In addition, initial humidity of the containment atmosphere is eliminated as a parameter. A value of 0% has been used for the peak LOCA and MSLB accident containment pressure and temperature calculations. This is conservative since it maximizes the mass of air in the containment, which maximizes the peak calculated containment accident pressure. The licensee has reduced the initial containment pressure for acceptable operation from 16.0 psia to 15.5 psia.

Since the results of the accident calculations, performed with these initial pressure and temperature conditions satisfy all regulatory criteria, the staff finds this change to be acceptable.

3.1.3 TS 6.15

TS 6.15, "Containment Leakage Rate Testing Program," is revised to reflect an increased calculated peak containment pressure, P_a, as a result of a LOCA. Since this pressure is slightly above the peak calculated pressure, and since the licensee must use this pressure in containment leakage rate testing, the staff finds this value acceptable.

3.1.4 USQ

The staff's review of the USQ associated with increasing the containment design pressure from 54 psig to 59 psig is discussed below.

The containment building is a prestressed concrete building consisting of a flat circular basemat, a right circular cylinder, and a sphere-torus dome. The containment has a height of 211'-9" with an internal cylinder radius of 58'-0" and an internal dome radius of 87'-6".

The containment walls are reinforced with steel reinforcing bars and post-tensioned with hoop and vertical tendons. The hoop tendons span 240 degrees around the containment building and are alternately anchored at each end to two of the three equally spaced vertical buttresses. The vertical tendons are seated and shimmed at the top of the ring girder and anchored at the bottom of the base slab in the tendon gallery.

The licensee analyzed the containment building for the faulted design condition which includes 59 psig pressure in combination with other design loads. The analysis was performed using a computer program, named Bechtel Structural Analysis Program (BSAP), which was developed by Bechtel Power Corporation. The BSAP employed the American Society of Mechanical Engineers (ASME), Section III, Division 2, 1975 design code. The containment building was originally designed in accordance with the ACI 318-1963 concrete code. The licensee stated that allowable stresses of the ACI 318-1963 concrete code were compared to the ASME, Section III, Division 2, 1975 code, and that the latter were found to be more stringent than the former. The staff considers such a comparison of design criteria acceptable.

The licensee stated that during construction of the containment building, nine surveillance tendons (three dome, three hoop, and three vertical) were installed but not credited in the original analysis, and that an additional nine tendons were installed (six hoop and three vertical) but were also not credited in the original analysis toward meeting the minimum prestress levels. These 18 tendons were credited in the re-analysis to meet the required prestress levels for the containment uprate. The staff considers the licensee's new analysis that accounts for the 18 tendons reasonable because they physically exist, and contribute to the load carrying capacity of the containment building.

The concrete cylinder compressive strengths at 90 days were consistently above 6700 psi. The new analysis used 5750 psi for concrete strength. The staff considers that the concrete strength used for the new analysis is conservative.

The new analysis used a concrete creep value from the twenty years of tendon surveillance performed on the Arkansas Nuclear One, Unit No. 1 (ANO-1) containment building. Surveillance data from the ANO-1 containment building was used, since the ANO-2 containment building had not had tendon surveillance, due to the similarity of the two containments in dimension and in construction material. The staff considers the concrete creep value used for the new analysis reasonable because it was taken from the actual surveillance data of the ANO-1 containment building, which is similar to the ANO-2 containment building in both structural and environmental conditions.

The licensee stated in its November 3, 1999, submittal that the new analysis had verified that the original design criteria were satisfied for the increased containment design pressure of 59 psig. In addition, the staff finds that the licensee had credited correct numbers of prestressing tendons, used a conservative concrete compressive strength, and selected a reasonable creep value for the new analysis.

The licensee states that it would perform a structural integrity test (SIT) at 1.15 times the new design pressure prior to resuming plant operation following replacement of the SGs, and that the SIT will assure that measured responses of the containment building are within the limits predicted by analyses. The staff accepts the licensee's proposed SIT because it meets the 1998 ASME, Section III, Division 2, Code requirements.

Based on the above review, the staff finds that the licensee's analysis methods, assumptions used for the analyses, and conclusions are based on sound structural engineering principles and the ASME code requirements that are acceptable to the staff.

3.2 LOCA Analysis

The LOCA causes a rapid depressurization of the reactor coolant system (RCS) due to the discharge of the reactor coolant into the containment, resulting in a sharp increase in the temperature and pressure of the containment atmosphere. The licensee modeled the RCS response and the containment response to the LOCA with different computer codes which account for the different phenomena which are important in the blowdown, refill, reflood, and post-reflood phases. In addition, the licensee assumed breaks at various locations in the RCS and different single failures to ensure that the most conservative case is analyzed. For peak containment pressure and temperature calculations, the licensee assumed breaks to occur in the hot leg, the pump suction leg, and the pump discharge leg of the RCS.

During the first phase of the LOCA, the blowdown, the RCS inventory is expelled into the containment. The larger primary system inventory of the RSGs affects the hot leg breaks. The licensee calculated the mass and energy of the reactor coolant discharged into the containment using the CEFLASH4A computer code. This code is used for 10 CFR 50.46 LOCA calculations. Application of this computer code for containment mass and energy release calculations has previously been approved by the staff. The licensee's November 3, 1999, submittal describes assumptions included in the calculations which increase the energy discharge into the containment. These include assuming that the heat transfer from the fuel rods to the coolant is always by nucleate boiling. In addition, swelling and rupture of the fuel rods is not permitted, even if it would normally be predicted to occur since neglecting this effect enhances heat transfer from the fuel rods. The refill phase is conservatively omitted for the containment calculations since this brings the injected water in contact with the fuel rods sooner, generating more steam which is subsequently discharged to the containment. The reflood phase is modeled with the FLOOD3 computer code. The decay heat modeled in the FLOOD3 computer code was calculated using the American National Standards Institute (ANSI)/American Nuclear Society (ANS) 5.1-1979 decay heat standard with a 2 standard deviation uncertainty. The application of this computer code and the decay heat standard have also been previously approved by the staff.

For the long-term cool-down phase, the mass and energy release and the containment response are calculated using the COPATTA computer code. The CONTRANS computer code calculated heat addition from the RCS loop and the SGs starting from the primary fluid temperature at the end of the post-reflood period for cold leg breaks. Both of these computer codes have been previously accepted by the staff for containment licensing calculations.

The licensee states that the containment analyses are done modeling the time-dependent behavior of the service water system which cools (among other things) the containment cooling system. Safety Analysis Report (SAR) Figure 6.2-3B gives the emergency cooling pond temperature as a function of time. This figure is identical to Figure 9.2-20 of the ANO-2 SAR. By supplemental letter dated August 16, 2000, the licensee confirmed that this figure remains bounding for the revised containment calculations reported in the November 3, 1999, application and June 29, 2000, supplement, even though these calculations were performed at a 7.5% higher power level. The licensee provided several reasons for this, including the use of a less conservative decay heat correlation. The licensee proposes to use the decay heat correlation of NRC Branch Technical Position 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling." This is conservative and acceptable.

The licensee determined that the worst single failure for the LOCA was loss of an emergency diesel generator which results in less water added to the RCS due to the loss of a high pressure emergency core cooling system (ECCS) pump and a low head ECCS pump. A loss of a containment cooling system train and a containment spray train will also occur.

The licensee has used NRC-approved calculational methods to perform the LOCA calculations. In addition, the staff has performed audit calculations (discussed below) with results similar to those performed by the licensee. For these reasons, the staff finds the licensee's LOCA calculations to be acceptable.

3.3 MSLB Analysis

The MSLB analysis calculates the mass and energy of steam discharged into the containment as a result of a ruptured main steam line. Using the results of this calculation, the peak containment pressure and temperature are calculated. The mass and energy analysis is calculated using the same methods used in a license amendment previously approved by the staff (Amendment 189, dated March 12, 1998).

The licensee credits new actuations initiated on the containment pressure high-high signal which isolate the main steam and feedwater flow, thus limiting the mass and energy addition to the containment. The licensee also credits the containment high pressure reactor trip, which occurs sooner than the low SG pressure reactor trip which was previously credited to limit the energy available for transfer to the containment.

The licensee used the RELAP MOD3 computer code to calculate the contribution of the main feedwater, including flashing, to the ruptured and intact SGs. The code simulated the ANO-2 main feedwater trains, including the main feedwater, condensate, and heater drain pumps; various valves; and feedwater heaters. No credit was taken for closing the feedwater regulating valves to stop feedwater flow. The transient pressures of the intact and ruptured SGs were then input to the RELAP5 MOD3 calculations. The feedwater flow rate and enthalpy to each

SG were input to the SGNIII computer code. The SGNIII code calculates the mass and energy release to the containment, which is used by the COPATTA containment computer code. COPATTA calculates the peak containment temperature and pressure. The SGNIII computer code has been approved for use in previous analyses by this licensee and others.

For the MSLB, the worst single active failure was the loss of a containment spray train at 0% power. The licensee assumed that offsite power is available for the containment peak pressure and temperature MSLB analysis. This maximizes the amount of heat transferred from the RCS to the SG and, consequently, maximizes the mass and energy release to the containment. Its effect is greater than that due to delay of the containment heat removal systems, which would result from assuming a loss of offsite power.

The licensee's June 29, 2000, supplemental letter informed the staff that credit for revaporization of condensate formed on the structural heat sinks during periods of superheat following a MSLB accident was assumed rather than transferring the condensate directly to the sump, which is the normal path in COPATTA. The licensee stated that this resulted in a decrease in the peak containment temperature, but did not affect the peak containment pressure. This is in accordance with the staff guidance in NUREG-0588, Appendix B, and is acceptable to the staff.

The licensee includes the effect of entrainment or carryover in the calculated mass and energy release to the containment by reducing the flow area of the break until no entrainment occurs. This is consistent with the approach used by the licensee and previously reported in the original ANO-2 SAR. The ANO-2 SAR states that the no-moisture carryover case at a reduced flow area produces the peak containment temperature and pressure for a given power level relative to the full flow area case with entrainment. Examples are given in ANO-2 SAR Table 6.2-12A.

The licensee has used NRC-approved calculational methods to perform the MSLB accident calculations. In addition, the staff has performed audit calculations (discussed below) that indicate that the licensee's calculations are conservative. For these reasons, the staff finds the licensee's MSLB accident calculations to be acceptable.

3.4 Audit Calculations

As part of the review of this ANO-2 license amendment request, the staff requested Los Alamos National Laboratory (LANL) to perform confirmatory (audit) calculations of peak containment pressure and temperature for the postulated LOCA and MSLB accident.

A model of the ANO-2 containment was developed based on information provided in the ANO-2 SAR and the license amendment request submitted by the licensee's November 3 1999, application and June 29, 2000, supplemental letter. The MELCOR computer code[2] was used to perform these calculations. Numerous sensitivity analyses were carried out (a) to select a set of key modeling parameters that impact model predictions of the containment pressure and

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[2]Recent comparative analysis performed by Sandia National Laboratories has shown that MELCOR and CONTAIN computer codes generate sufficiently similar results for the current objectives. MELCOR provided more flexibility to handle the engineered safety features, and hence was selected in this study.

temperature, and (b) to quantify their impact. The sensitivity analysis results were used to provide upper bound, baseline, and lower bound values for the peak containment pressure and temperature.

The results for the LOCA analysis showed reasonable agreement with those of the licensee. Trends were similar and the timing of key events, such as the receipt of containment cooling system and containment spray actuation signals, and the time of peak containment pressure confirm the results of the licensee's calculations. For the MSLB accident, the staff calculated peak containment pressure and peak containment temperature values significantly less than those calculated by the licensee. In a July 27, 2000, telephone conversation, licensee representatives suggested that these differences might be due to the conservative modeling of containment spray efficiency in the COPATTA code (described in Section 6.2.1.3.2.5 of the current ANO-2 SAR). A modification of the MELCOR containment spray model to make the containment spray less effective confirmed this as the likely cause of the difference. In any case, the LANL calculations confirm that the containment design pressure and temperature limits will not be exceeded during a design-basis LOCA or MSLB accident. It is likely that actual peak pressure and temperature would be significantly lower than the design limits.

These calculations are reported in "Confirmatory Analysis: ANO-2 Design-Basis Accident Containment Response," LA-UR-3701, August 2000. This report is available in the NRC Agencywide Documents Access and Management System (ADAMS), accession number ML003745334, via the NRC website (www.nrc.gov).

3.5 Environmental Qualification (EQ)

The licensee's re-analyses for these RSGs found increased containment pressure during accident conditions. This section discusses the licensee's evaluation of the impact related to these increased conditions on the EQ of components located inside containment.

With regard to EQ, the licensee indicated, by supplemental letter dated June 9, 2000, that they evaluated equipment required to operate during accident conditions using new limiting LOCA peak conditions of 57.6 psig and 285°F. The licensee indicated that (1) peak test conditions (utilized to environmentally qualify equipment for temperature and pressure) envelops the new limiting LOCA peak conditions; (2) three equipment types have peak test pressure conditions that envelop the new limiting peak pressure conditions, but not with the margins suggested by Institute of Electrical and Electronics Engineers (IEEE) 323-1974 (10% for pressure); and (3) equipment based on testing of an identical item under identical or similar conditions with supporting analysis (which is an accepted method permitted by paragraph (f) of 10 CFR 50.49) demonstrates that equipment remains qualified for the new limiting LOCA conditions.

In regard to the three equipment types that were not tested with 10% margin for pressure, the licensee, by supplemental letter dated June 9, 2000, supported their conclusion that these three equipment types meet the margin requirements of 10 CFR 50.49 and remain qualified for the new limiting LOCA conditions, as summarized below:

1) Incore thermocouple cable/connector assembly

This equipment was tested to a peak pressure condition of 59 psig for greater than 1200 seconds, which does not meet the suggested 10% pressure margin of 63.36 psig (57.6 psig plus 10% margin). This peak test pressure of 59 psig for 1200 seconds, when compared to the required design conditions of 57.6 psig peak transient pressure for approximately 100 seconds, provides adequate margin; thus, pressure qualification is justified.

2) American Insulated Wire (AIW)

AIW was tested to a pressure of 60 psig, which bounds the required design conditions of 57.6 psig but does not meet the suggested 10% pressure margin of 63.36 psig. Pressure has no degrading mechanism on cable; therefore, not meeting the suggested margin poses no qualification concern for cable. Further, similar AIW cable has been tested to 86 psig and demonstrated full qualification. Thus, pressure qualification is justified.

3) Amphenol electrical penetrations

The penetrations were tested to a pressure of 63 psig, which bounds the required design conditions of 57.6 psig; however, they do not meet the suggested 10% pressure margin of 63.36 psig. This equipment is being modified with new module seals. The new module seals have been tested to a peak pressure condition of 69.76 psig. Thus, after modification, penetrations bound the suggested 10% margin of 63.36 psig and are considered qualified for pressure.

Based on the licensee's evaluation and supporting analysis, the staff finds that equipment remains qualified for the new limiting LOCA peak conditions of 57.6 psig and 285°F. With regard to these new accident conditions, the staff concludes that equipment meets the qualification requirements of 10 CFR 50.49 and can be considered acceptably qualified.

In regard to the use of 285°F as the new peak temperature for equipment qualification, the licensee indicated, by supplemental letter dated June 9, 2000, that their LOCA containment design basis accident analysis has been revised to include a revaporization assumption. With this added assumption, the 285°F temperature value became the new peak temperature for equipment qualification.

During discussions with the licensee, the licensee indicated that the EQ test profile did not envelop the revised limiting LOCA temperature profile for post-LOCA conditions. In justification for this case, the licensee compared equipment degradation caused by the EQ test profile to degradation that would be caused by the new accident profile using the Arrhenius calculation. The licensee found greater (or equivalent) calculated degradation from the EQ test profile than from the new accident profile and thus concluded equipment continues to be qualified to the new post-accident temperature conditions. Based on their being greater (or equivalent) degradation, the staff concluded that there is reasonable assurance that components will continue to have the required capability to perform their safety function when needed and finds that equipment continues to be acceptably qualified for the new temperature conditions.

With regard to pressure exceeding EQ limits during post-LOCA conditions, the licensee by supplemental letter dated June 9, 2000, provided the following justification for their conclusion that equipment remains qualified:

> "ANO's [Arkansas Nuclear One's] EQ program established a process to evaluate the equipment's tested condition against the plant-required conditions. ANO accident evaluations include a graphical presentation of the accident test temperature and pressure profile versus the accident required temperature and pressure profile for each type of EQ equipment located inside containment. With respect to pressure, the test and required profiles are visually compared. The fundamental concern during the accident period is the long-term effects of elevated temperature conditions. Since thermal degradation is basically an oxidation reaction, the chemical reaction would not be significantly affected by a slight pressure during the long-term accident period (i.e., the temperature profile equivalency evaluation is not affected by this slight, long-term pressure). The challenge to equipment from the accident pressure is maintaining the equipment's integrity (e.g., to prevent pressure from crushing the enclosure or seals/gaskets leaking, thus allowing the external environment into the enclosure). The ability of enclosures to tolerate differential pressures is most challenged during the initial high temperature, high pressure transient conditions. The long-term accident pressure is significantly lower than during the accident transients. Physical failures related to pressure classically do not occur during the long-term, post-accident period. Adverse seal/gasket leakage effects, if any, would become evident during the higher-pressure transient period. Many equipment types are not functionally affected by the external environment entering the enclosure (e.g., Limitorque motor operated valve actuators) and as such are not required to be sealed against the pressure.
>
> Therefore, total graphical envelopment by the test pressure profile over the required pressure profile for the entire duration is not a significant concern. The main concern is the highest pressure that the equipment will experience during an accident. Therefore, the testing is evaluated to ensure that the equipment is subjected to at least the required peak pressure conditions."

Based on this justification, the staff agrees that bounding the peak accident pressure with margin as part of EQ type-testing adequately demonstrates qualification. The staff therefore concludes that equipment meets the qualification requirements of 10 CFR 50.49 and can be considered to remain acceptably qualified for the new containment accident conditions.

For the reasons described above, the staff concludes that electrical equipment continue to meet the requirements of 10 CFR 50.49 and remain qualified for the new containment accident conditions.

3.6 Containment Spray Pump Net Positive Suction Head (NPSH)

The licensee indicated that the NPSH of the containment spray pumps will not be adversely affected by the calculated conditions within containment, including an increase in the temperature of the spilled water on the containment floor. The licensee indicated that the reduction in static head due to the higher water temperature will be offset by the lower density and, consequently, lower head loss between the sump and the pump suction. The larger water

inventory in the RSGs will also result in a higher water level in the containment during recirculation, which increases available NPSH. No credit was taken for the containment pressure in calculating available NPSH. For these reasons, the staff finds the NPSH for the containment spray pumps acceptable.

3.7 RCS Leakage Detection

The licensee plans to apply leak-before-break technology to primary loop piping as part of the SG replacement effort. In particular, the licensee's November 3, 1999, submittal discusses the application of leak-before-break in determining containment subcompartment loads on the primary shield walls around the reactor vessel and the secondary shield walls around the SGs to determine that they are adequate for the increased pressure. As part of a previous application of leak-before-break technology, a September 16, 1994, letter from the licensee provided the staff with a comparison of the capability of the ANO-2 RCS leak detection system with the guidelines of Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." In particular, the licensee stated that the containment sump level monitoring system is one of the primary leak detection systems. Leakage detection systems are discussed in Section 5.2.7.1 of the ANO-2 SAR, and operability and surveillance requirements for these systems are specified in TS 3/4.4.6. The licensee stated that these systems are capable of detecting a leak rate of one gpm.

Subsequently, an August 11, 1998, letter from the licensee to the NRC informed the staff that, "...in a certain circumstance, a one gpm increase in leakage into the sump could require up to 70 minutes to be detected." This exceeds the criterion of RG 1.45 that a leak of one gpm be detected within one hour. There are other leak detection instruments which could also detect a leak rate of one gpm from the RCS. The ANO-2 TSs credit, in addition to sump level, the containment atmosphere particulate radioactivity monitoring system and the containment atmosphere gaseous radioactivity monitoring system. According to the licensee's September 16, 1994, letter, the latter two methods also may not detect RCS leakage within one hour, depending, for example, on the activity in the coolant.

The staff considers this change to be acceptable since, in the context of leak-before-break technology and also from the perspective of overall plant risk, the difference between detecting a one gpm leak in 60 minutes versus 70 minutes by means of the sump level monitoring system is not significant.

3.8 Adequacy of Existing Containment Wide Range Pressure Transmitters

The current transmitters have a calibrated range of 0-210 psia. RG 1.97 requires that these transmitters should have a range of three times the containment absolute design pressure. So, the current transmitters meet the requirements of RG 1.97 for the existing SGs for which the containment design pressure is 54 psig or 69 psia. For the new SGs, the containment design pressure is 59 psig or 74 psia, and the transmitter range ought to be 0-222 psia. Therefore, these transmitters fall 12 psia below the required range. The licensee proposed to use the existing transmitters because these transmitters are marginally short of the required range and the transmitter range is significantly above the ultimate pressure capability of the containment building. In response to staff inquiry, the licensee confirmed by supplemental letter dated April 4, 2000, that these transmitters are used only for indication and recording purposes and

not for any automatic protective function. Considering above justifications, the staff finds that the current transmitters meet the intent of RG 1.97 and need not be replaced. However, in case replacement is needed for these items for any other reason, the licensee should replace these transmitters with a wider range of 0-222 psia.

3.9 Evaluation Summary

The staff finds the licensee's proposed TS changes and the USQ resolution concerning the increased containment design pressure, to be acceptable, based on a review of the licensee's submittals and audit calculations of the peak containment pressure and temperature due to a postulated LOCA and a postulated MSLB accident.

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 (65 FR 9006, dated February 23, 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.

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.

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