September 19, 1990

Docket No. 50-331

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Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
Iowa Electric Light and Power Company
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

SUBJECT: AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-49 (TAC NO. 69009)

The Commission has issued the enclosed Amendment No. 169 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center (DAEC). This amendment consists of changes to the Technical Specifications (TS) in response to your application dated July 27, 1988, as revised June 29, 1990.

The amendment revises the DAEC TSs to conform with the guidance of NRC Generic Letter (GL) 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." Additional changes updating schedules for the 10-year inservice inspection and testing programs and other clarifications are included.

Your GL 88-01 inspection program was previously approved by letter dated May 31, 1990. In that letter, the staff indicated that the 81 Category G welds in the portion of Reactor Water Cleanup system piping replaced during the current refueling outage would not be required to be inspected as part of your GL 88-01 program. For clarification, those welds do not require inspection during this refueling outage, but should be retained in the inspection program under the appropriate category.

Sincerely,

James R. Hall, Project Manager
Project Directorate III-3
Division of Reactor Projects - III, IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 169 to License No. DPR-49
2. Safety Evaluation

cc w/enclosures: See next page

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

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 169
License No. DPR-49

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

   A. The application for amendment by Iowa Electric Light and Power Company, et al., dated July 27, 1988, as revised June 29, 1990, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 169, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John N. Hannon

John N. Hannon, Director
Project Directorate III-3
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of issuance: September 19, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 169

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

3.2-20

3.2-45

3.6-5

3.6-8

3.6-9

3.6-25

3.6-26

3.6-36

3.6-37

TABLE 3.2-E

INSTRUMENTATION THAT MONITORS DRYWELL LEAK DETECTION

Minimum No. of Operable Instrument Channels	Instrument	No. of Instrument Channels Provided by Design	Action
1	Sump System (1)	6	(3)
1	Air Sampling System (2)	6	(3)

NOTES FOR TABLE 3.2-E

(1) The Sump System is comprised of the Equipment Drain Sump and Floor Drain Sump Sub-systems.

The Equipment Drain Sump Sub-system consists of one Equipment Drain Sump Flow Integrator and two Equipment Drain Sump Flow Timers. The Floor Drain Sump Sub-system likewise consists of one Floor Drain Sump Flow Integrator and two Floor Drain Sump Flow Timers. The Sump Sub-system is operable when any one of these six devices operable.

(2) The Air Sampling System provides a backup system to the Sump System.

Action for Table 3.2-E

(3) See Specification 3.6.C.

timer is set to annunciate before the values specified in Specification 3.6.C are exceeded. An air sampling system is also provided, as a backup to the sump system, to detect leakage inside the primary containment.

For each parameter monitored, as listed in Table 3.2.F, there are two (2) channels of instrumentation. By comparing readings between the two (2) channels, a near continuous surveillance of instrument performance is available. Any deviation in readings will initiate an early recalibration, thereby maintaining the quality of the instrument readings.

On July 26, 1984 the NRC published their final rule on Anticipated Transients Without Scram (ATWS), (10 CFR § 50.62). This rule requires all BWR's to make certain plant modifications to mitigate the consequences of the unlikely occurrence of a failure to scram during an anticipated operational transient. The bases for these modifications are described in NEDE-31096-P-A, "Anticipated Transients Without Scram; Response to NRC ATWS Rule, 10 CFR 50.62," December 1985. The Standby Liquid Control System (SLCS) was modified for two-pump operation to provide the minimum required flowrate and boron concentration required by the ATWS rule (see section 3.4 Bases). The existing ATWS Recirculation Pump Trip (RPT) was modified from a one-out-of-two-once logic to trip each recirc. pump to a two-out-of-two-once logic to trip both recirc. pumps ("Monticello" design). This logic will also initiate the Alternate Rod Insertion (ARI) system, which actuates solenoid valves that bleed the air off the scram air header, causing the control rods to insert. The instrument setpoints are chosen such that the normal reactor protection system (RPS) scram setpoints for reactor high pressure or low water level will be exceeded before the ATWS RPT/ARI setpoints are reached. Because ATWS is considered a very low probability event and is outside the normal design basis for the DAEC, the surveillance frequencies and LCO requirements are less stringent than for safety-related instrumentation.

The End-of-Cycle (EOC) recirculation pump trip was added to the plant to improve the operating margin to fuel thermal limits, in particular Minimum Critical Power Ratio (MCPR). The EOC-RPT trips the recirc. pumps to lessen the severity of the power increases caused by either a closure of turbine

LIMITING CONDITION FOR OPERATION

C. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment shall be limited to:

a. 5 gpm unidentified leakage.

b. 2 gpm increase in unidentified leakage within a 24 hr. period.

c. 25 gpm total leakage.

2. The sump system shall be operable any time irradiated fuel is in the vessel and reactor coolant temperature is above 212°F. From and after the date that the sump system is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 24 hours unless the system is made operable sooner, provided the air sampling system is operable.

3. If the conditions in 1 or 2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

D. Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than atmospheric and temperature greater than 212°F, both safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2.

SURVEILLANCE REQUIREMENT

C. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump system and recorded at least once every 8 hours.

2. The air sampling system shall be checked and recorded at least once every 8 hours.

D. Safety and Relief Valves

1. At least one safety valve and 3 relief valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles.

The setpoint of the safety valves shall be as specified in Specification 2.2.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
3. Following 1-pump operation, the discharge valve of the lower speed pump may not be opened unless the speed of the faster pump is less than 50% of its rated speed.	
G. Structural Integrity	G. Structural Integrity
The structural integrity of the pressure boundaries shall be maintained at the level required by the original acceptance standard throughout the life of the plant.	1. In-service inspection of ASME Code Class I, Class II and Class III components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
	a. The second 10 year interval for the inservice inspection program described above commenced on November 1, 1985.
	2. In-service testing of ASME Code Class I, Class II and Class III pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).
	a. The second 10-year interval for the inservice testing program described above commenced on February 1, 1985.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	3. The inservice inspection program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods and personnel, and sample expansion included in this generic letter.

establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propagation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

Identified and unidentified leakage are defined in the DAEC Updated FSAR, Section 5.2.5.2.2. Total leakage is defined as the sum of identified and unidentified leakage.

The capacity of the drywell floor sump pumps is 50 gpm and the capacity of the drywell equipment sump pumps is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with margin.

DAEC surveillance procedures require both identified and unidentified leakage to be determined at approximately 4 hour intervals. Should leakage exceed the allowed limits, control room alarms actuated by the equipment drain sump and floor drain sump pump timers are provided to indicate this condition, thus, continuous leakage detection capability is provided by design.

The requirement that an increase in unidentified leakage shall not exceed 2 gpm in a 24 hour period is based on maintaining the ability to detect small leaks in a reasonably short time such that corrective action can be initiated. However, during reactor startup and ascension to normal operating pressure, leakage should be closely monitored until normal operating pressure is achieved and a "baseline" leakage rate can be established to which any leakage increase can be compared.

The primary containment atmosphere radioactivity detector provides a sensitive and rapid indication of increased nuclear system leakage. The primary containment environment is continuously sampled from one of three locations which are chosen to provide both a representative gas mixture and an indication of the location of the leakage.

The sample air undergoes three separate processes in which the radioactive noble gas, halogen, and particulate contents are determined. This system is thus a three channel monitoring system. The processed air is returned to the drywell.

The primary containment atmosphere radioactivity detector serves as a sensitive, reliable backup to the other metnods of leak detection. It is anticipated that the particulate detector will be the primary indication of leakage, with the halogen and noble gas detectors serving as indication of the primary containment environment if primary containment venting is required. These detectors in conjunction with an isotopic analysis can be used to indicate whether the detected leak is from a steam or water system. This system is not capable of accurately quantifying coolant leakage rates. Because the Air Sampling system is not capable of determining leak rate, it is considered a backup system to the sump system, and no LCO is associated with it. It is intended to be a compensatory measure used when the sump system is inoperable.

The first 10-year interval for inservice inspections in accordance with the ASME Boiler and Pressure Vessel Code, Section XI commenced on February 1, 1975. This interval was extended for 9 months because of a 9 month outage for replacement of recirculation system inlet nozzle safe-ends in 1978-79. Therefore, the first 10-year interval ended on October 31, 1985.

The second 10-year interval for inservice inspections commenced on November 1, 1985 and is scheduled to end on October 31, 1995. The second 10-year inspection program addresses the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1981, subject to the limitations and modifications as stated in 10 CFR 50.55a.

Visual inspections for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsered by the NRC and is delineated by Section XI of the ASME Code. These studies show that it requires thousands of stress cycles at stresses beyond those expected to occur in a reactor system to propagate a crack. The test frequency established is at intervals such that in comparison to study results, only a small number of stress cycles, at values below limits will occur. On this basis, it is considered that the test frequencies are adequate.

The type of inspection planned for each component depends on location, accessibility, and type of expected defect. Direct visual examination is proposed wherever possible since it is fast and reliable. Surface inspections are planned where practical, and where added sensitivity is required. Ultrasonic testing or radiography shall be used where defects can occur in concealed surfaces. Section 5.2.4 of the Updated FSAR provides details of the inservice inspection program.

Starting with the Cycle 9/10 Refueling Outage, an augmented inspection program was implemented to address concerns relating to Intergranular Stress Corrosion Cracking (IGSCC) in reactor coolant piping made of austenitic stainless steel. The augmented inspection program conforms to the NRC staff's positions set forth in Generic Letter 88-01 and NUREG-0313, Revision 2 for inspection schedule, inspection methods and personnel, and inspection sample expansion.

The first 10-year interval for inservice testing of pumps and valves in accordance with the ASME Code, Section XI commenced on February 1, 1975 and ended on January 31, 1985. The second 10-year inservice testing interval commenced on February 1, 1985 and is scheduled to end on January 31, 1995. The second 10-year testing program addresses the requirements of the ASME Code, Section XI, 1980 Edition with Addenda through Winter 1981, subject to the limitations and modifications of 10 CFR 50.55a. Section 3.9.6 of the Updated FSAR describes the inservice testing program.

3.6.H & 4.6.H BASES:

Shock Suppressors (Snubbers)

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or other severe transient, while accommodating normal thermal motion during system startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of damage to piping as a result of a seismic or other event initiating dynamic loads or, in the case of a frozen snubber, exceeding allowable stress limits during system thermal transients. It is therefore required that all snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 169 TO FACILITY OPERATING LICENSE NO. DPR-49

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated July 27, 1988, Iowa Electric Light and Power Company (the licensee) requested changes to the Duane Arnold Energy Center (DAEC) Technical Specifications (TSs). The proposed changes were submitted in response to NRC Generic Letter (GL) 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," and its enclosure, NUREG-0313, Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

The NRC staff approved the licensee's proposed inspection program and response to GL 88-01 in a letter dated May 31, 1990. That letter did not address the proposed TS changes that are reviewed in this Safety Evaluation, except to identify that the licensee's proposed frequency of monitoring reactor coolant system (RCS) leakage once per 24 hours was unacceptable. The May 31, 1990 letter also described a change in the NRC staff position regarding the frequency of leakage monitoring. The staff position in GL 88-01 had specified that leakage monitoring of sump levels using fixed-measurement-interval methods should be conducted at 4-hour intervals or less. The staff has subsequently relaxed the specified frequency to once every 8 hours, due to the unnecessary administrative hardship imposed by the 4-hour interval. The licensee revised its request of July 27, 1988 by letter dated June 29, 1990, to conform with the staff's current position on RCS leakage monitoring.

The proposed changes clarify Table 3.2-E, "Instrumentation that Monitors Drywell Leak Detection," revise TSs 3.6.C and 4.6.C, and add TS 4.6.G.3 to conform with the staff positions of GL 88-01, as revised. In addition, the proposed changes revise TSs 4.6.G.1.a and 4.6.G.2.a to specify the beginning dates for the second 10-year inservice inspection (ISI) and inservice testing (IST) programs. TS 4.6.G.3 is deleted, as the first 10-year ISI and IST intervals have been completed, and TS 4.6.G.4 is also deleted, as the interim ISI program for the recirculation system inlet nozzle safe-ends has been completed and future inspections will be included in the scope of the GL 88-01 program. The associated Bases for the TSs are also revised to reflect the proposed changes.

2.0 EVALUATION

The licensee has proposed revisions to Table 3.2-E, "Instrumentation that Monitors Drywell Leak Detection," and to TS 3.6.C.2, to clarify the operability requirements for those systems used for drywell leak detection. The licensee provided additional information on these systems in a letter dated April 24, 1989.

The primary system used for the detection of drywell leakage at the DAEC is the Sump system, consisting of the Equipment Drain Sump subsystem and the Floor Drain Sump subsystem. Each subsystem is comprised of a flow integrator, a sump pump run timer, and a sump fill timer. Each of these devices can be used by plant operators to calculate drywell leakage rates. The flow integrators receive signals from flow transmitters located in the discharge piping of both sumps, and calculate the total amount of fluid discharged from each sump. This information is used by the operators to calculate the drywell leakage rates and record them at 4-hour intervals in accordance with plant procedures. The two sump pump run timers measure the length of time each pump runs, from the point that a high sump level starts the pump, until it shuts off automatically upon reaching the low level setpoint. For a given pump flow rate, the time the pump is running corresponds to a set drywell leakage rate. Therefore, if the pump is running for too long, a high drywell leakage rate is indicated in the control room. The sump fill timers measure the time between successive pump starts. These timers are set to correspond to the time period between pump starts for an established pump flow rate and specified drywell leakage rate. If the pump restarts prior to reaching the set time interval, then drywell leakage is greater than the setpoint and an annunciator is activated in the control room.

Any one of these six instruments is sufficient to detect increased drywell leakage. Identified leakage, which is composed of normal seal and valve packing leakage, is collected in the Equipment Drain Sump, while unidentified leakage, composed of all other leakage from the reactor primary system, is collected in the Floor Drain Sump. The two sumps are adjacent to each other, located beneath the reactor, inside the reactor vessel pedestal. If all three leak detection instruments in one sump were inoperable, the pumps in that sump would not start automatically and the sump would eventually overflow into the adjacent sump. Based on the 850-gallon capacity of each sump (and the 200 gallon low-level setpoint), at a drywell leakage rate of 5 gpm, one sump would begin to overflow to the other in just over 2 hours. At a leakage rate of 2 gpm, the time would be roughly 5½ hours.

The Floor Drain Sump instrumentation is set to detect unidentified drywell leakage, while the Equipment Drain Sump instrumentation is set to detect identified leakage. In the event that all three Floor Drain sump components are inoperable, the Equipment Drain Sump timers would be recalibrated to the lower setpoint for detection of identified leakage. Therefore, if any one of the six instruments of the Sump system was operable, plant operators could detect high drywell leakage and take appropriate actions in a reasonable amount of time.

The Air Sampling System provides backup capability to detect drywell leakage, by monitoring increases in the radioactivity in the drywell atmosphere. However, this system is not capable of quantifying drywell leakage and is therefore only intended to be used when the Sump system is inoperable.

Proposed TS 3.6.C.2 specifies that the Sump system shall be operable any time irradiated fuel is in the vessel and reactor coolant temperature is above 212°F. If the Sump system is inoperable, continued reactor operation is permissible for 24 hours only if the Air Sampling System is operable; otherwise, the reactor shall be in the Cold Shutdown Condition within 24 hours. This revised specification has a 24-hour limiting condition for operation (LCO) instead of the current 7-day LCO; however, the current LCO requires that the Air Sampling System be operable in addition to the Sump system.

The NRC staff finds that the revised Table 3.2-E and revised TS 3.6.C.2 more accurately reflect the redundant design of the DAEC Sump system drywell leakage monitoring instrumentation and the backup function of the Air Sampling System, thereby providing more appropriate requirements for drywell leak detection. The Bases of pages 3.2-45 and 3.6-26 have also been revised to reflect these changes, which the staff finds acceptable.

Proposed TS 3.6.C.1 and 4.6.C.1 and 2 have been revised consistent with the staff's position in GL 88-01, as modified, in part, in the May 31, 1990 NRC letter to the licensee. TS 3.6.C.1.b adds a limit of 2 gpm increase in unidentifed leakage within a 24-hour period. If this additional limit is exceeded, the reactor shall be in a Cold Shutdown Condition within 24 hours, as required by TS 3.6.C.3. As a point of clarification, if a 2 gpm increase in unidentified leakage was observed in less than 24 hours, the limit would also be considered to be exceeded and the appropriate action required, consistent with the wording of GL 88-01. The revised Bases of page 3.6-25 provide additional clarification. Surveillance Requirements (SRs) 4.6.C.1 and 2 require the RCS leakage to be checked by the sump system and recorded once every 8 hours and the Air Sampling system to be checked and recorded once every 8 hours. These revised TSs are consistent with the current staff positions as described in GL 88-01 and the May 31, 1990 NRC letter; therefore they are acceptable.

SR 4.6.D.1 is revised to delete a footnote referring to a previous change. Deletion of the footnote is an editorial change that clarifies the requirement and is therefore acceptable.

Proposed SRs 4.6.G.1.a and 4.6.G.2.a and the associated Bases specify the starting dates for the second 10-year intervals for the ISI and IST programs, respectively. Also, SR 4.6.G.3. is deleted, removing extraneous information regarding the completed first 10-year intervals. These sections are added or deleted for clarification and do not alter any existing requirements; therefore, the staff finds these changes acceptable.

A new Surveillance Requirement 4.6.G.3 is proposed, in which the licensee commits to perform an inservice inspection program for piping identified in NRC GL 88-01, in accordance with the staff positions contained therein. The proposed SR is worded exactly the same as the sample specification provided in GL 88-01. Therefore, the staff finds the proposed SR acceptable.

Finally, SR 4.6.G.4 is deleted to remove the references to the interim testing program for the recirculation system inlet nozzle safe-ends. Inspection of these components will be continued at the same frequency within the scope of the licensee's GL 88-01 inspection program, as required by SR 4.6.G.3. Therefore, the deletion of SR 4.6.G.4 does not alter existing requirements and is acceptable to the staff.

In summary, the proposed TS changes conform with the staff positions of GL 88-01, as they commit the licensee to conduct an approved inservice inspection program for piping susceptible to intergranular stress corrosion cracking. The licensee's GL 88-01 inspection program for the DAEC was previously approved by the staff in a letter dated May 31, 1990.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: James R. Hall, NRR

Dated: September 19, 1990