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

April 10, 1980

Docket No. 50-331

Mr. Duane Arnold, President
Iowa Electric Light & Power Company
P. O. Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Arnold:

The Commission has issued the enclosed [redacted] Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications in response to your following applications: (1) August 30, 1977, as revised March 18, 1980, (2) October 22, 1979, as revised April 9, 1980, and (3) March 4, 1980.

This amendment incorporates provisions into the Technical Specifications for (1) modifications associated with degraded grid voltage protection, (2) installation of the end-of-cycle recirculation pump trip, and (3) modifications in conjunction with the Mark I Containment Long Term Program.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 58
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

cc:

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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

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 58
License No. DPR-49

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

   A. The applications for amendment by Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative (the licensee) dated August 30, 1977 (supplemented March 18, 1980), October 22, 1979 (supplemented April 9, 1980), and March 4, 1980, comply 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 applications, 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 Appendices A and B, as revised through Amendment No. 58, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 10, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 58

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 pages are identified by Amendment Number and contain vertical lines indicating the area of change.

3.2-4

3.2-14

3.2-15

3.2-23

3.2-26

3.2-34

3.7-14

3.7-41

3.8-2

*3.8-11

3.8-12

*No change. Provided for convenience

Recirculation Pump Trip

(ATWS)

The limiting conditions for operation for the instrumentation that trips the recirculation pumps as a means of limiting the consequences of a failure to scram during an anticipated transient are given in Table 3.2-G.

(EOC)

The limiting conditions for operation for the instrumentation that trips the recirculation pumps during turbine stop valve or control valve fast closure for transient margin improvement (especially for end of cycle) are given in Table 3.2-G.

G. Recirculation Pump Trip

Instrumentation and logic shall be functionally tested, calibrated, and response time tested as indicated on Table 4.2-G.

TABLE 3.2-B (Continued)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Number of Instrument Channels Provided by Design	Remarks
1	RCIC Leak Detection Time Delay	30 min.	2 Inst.	
2 (5)	HPCI Steam Line Low Pressure	$100 > P > 50$ psig (3)	4 Inst.	
2	HPCI Equipment Room High Ambient Temperature	$\leq$ 175 deg. F	4 Inst.	
2	HPCI Equipment Room High Diff. Temperature	$\leq \Delta 50$ deg. F (3)	4 Inst.	
1 per 4 kV Bus	4 kV Emergency Bus Undervoltage	$20 \leq V \leq 28$ Volts	2	1. Trips all loaded breakers 2. Fast transfer permissive 3. Dead bus start of diesel
1 per 4kV Bus	4 kV Emergency Bus Sequential Loading Relay	65% of Rated Voltage	2	Permits Sequencing of vital loads
2 per 4kV Bus	Emergency Transformer Undervoltage	65% of Rated Voltage	4	1. Trips emergency transformer feed to 4KV emergency bus 2. Fast transfer permissive
1 per 4 KV Bus (7)	4 KV Emergency Bus Degraded Voltage	$108 \leq V \leq 111$ Volts $8.0 \leq$ T.D. ≤ 8.5 Sec.	2 Matrices	1. Trips 4 KV emergency bus incoming breakers 2. Starts diesel 3. Permits sequencing of vital loads

NOTES FOR TABLE 3.2-B

1. Whenever any CSCS subsystem is required by Subsection 3.5 to be operable, there shall be two operable trip systems. If the first column cannot be met for one of the trip systems, that trip system shall be placed in the tripped condition or the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

2. Close isolation valves in RCIC subsystem.

3. Close isolation valves in HPCI subsystem.

4. Instrument setpoint corresponds to 18.5" above the top of active fuel.

5. HPCI has only one trip system for these sensors.

6. The relay drop-out voltage will be measured once per operating cycle and the data examined for evidence of relay deterioration.

7. Four undervoltage relays with integral timers per 4 KV bus. The relay output contacts are connected to form a one-out-of-two-twice coincident logic matrix. With one relay inoperable, operation may proceed provided that the inoperable relay is placed in the tripped condition within one hour.

TABLE 3.2-G

INSTRUMENTATION THAT INITIATES RECIRCULATION PUMP TRIP

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Number of Instrument Channels Provided By Design	Action
1	(ATWS) Reactor High Pressure	≤ 1120 psig	4	(2)
1	(ATWS) Reactor Low Water Level	≥ -38.5 in. indicated level	4	(2)
1	(EOC) RPT Logic	N/A	2	(3)
1	(EOC) RPT System (Response Time)	≤ *msec (4)	2	(3)

NOTES FOR TABLE 3.2-G

1. Whenever the reactor is in the RUN Mode, there shall be one operable trip system for each parameter for operating recirculation pump. If this cannot be met, the indicated action shall be taken.

2. Reduce power and place the mode selector switch in a mode other than the RUN Mode.

3. Two EOC RPT systems exist, either of which will trip both recirculation pumps. The systems will be individually functionally tested monthly. If the test period for one RPT system exceeds 2 consecutive hours, the system will be declared inoperable. If both RPT systems are inoperable or if 1 RPT system is inoperable for more than 72 consecutive hours, an orderly power reduction shall be initiated and the reactor power shall be less than 85% within 4 hours.

4. This response time is from initiation of turbine control valve fast closure to actuation of the breaker auxiliary contact.

*To be determined by testing after installation. (Value to be design requirement for breaker opening less difference between cycle time for loaded vs. unloaded breaker.)

TABLE 4.2-B

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CSCS

Instrument Channel	Instrument Functional Test (9)	Calibration Frequency (9)	Instrument Check
1) Reactor Water Level	(1)	Once/3 months	Once/day
2) Drywell Pressure	(1)	Once/3 months	None
3) Reactor Pressure	(1)	Once/3 months	None
4) Auto Sequencing Timers	N/A	Once/operating Cycle	None
5) ADS - LPCI or CS Pump Discharge Pressure Interlock	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	Not Applicable	None
7) Recirculation System d/p	(1)	Once/3 months	Once/day
8) Core Spray Sparger d/p	(1)	Once/3 months	Once/day
9) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
10) Steam Line High Temp (HPCI & RCIC)	(1)	Once/operating Cycle	Once/day
11) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None
12) HPCI Suction Source Levels	(1)	Once/3 months	None
13) a. 4KV Emergency Power System Voltage Relays	Once/operating Cycle	Once/operating cycle	None
b. 4KV Emergency Power System Voltage Relays (Degraded Voltage)	Once/month	Once/operating cycle	None
14) Instrument A.C. and battery bus undervoltage relays	(1)	Once/operating cycle	None

TABLE 4.2-G

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RECIRCULATION PUMP TRIP

(ATWS)

Instrument Channel	Instrument Functional Check	Calibration Frequency
Reactor High Pressure	Once/refueling cycle	Once/refueling cycle
Reactor Low Water Level	Once/refueling cycle	Once/refueling cycle

Logic System Function Test	Frequency
Recirculation Pump Trip	Once/refueling cycle

(EOC)

Instrument Channel	Functional Check	Calibration Frequency	Instrument Check	Response Time
RPT Initiate Logic	Once/Month	N/A	N/A	N/A
RPT System	Once/Operating Cycle	N/A	N/A	Once/Operating Cycle

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
must be taken out of power operation	functionally tested once per operating cycle in conjunction with specification 4.7.A.6.a. Should one of the two H_2 or O_2 analyzers serving the drywell or suppression pool be found inoperable, the remaining analyzer of the same type serving the same compartment shall be tested for operability once per week until the defective analyzer is made operable.
7. Drywell-Suppression Chamber Differential Pressure	7. Drywell-Suppression Chamber Differential Pressure
a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.10 psid except as specified in (1) and (2) below:	a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.
(1) Within the 24-hour period subsequent to placing the reactor in the Run Mode following a shutdown, the differential shall be established. The differential may be decreased to less than 1.10 psid 24 hours prior to a scheduled shutdown.	
(2) This differential may be decreased to less than 1.10 psid for a maximum of four hours during required operability testing of the HPCI system pump, the RCIC system pump, the drywell-pressure suppression chamber vacuum breakers, and the suppression chamber to reactor building vacuum breakers.	
b. If the differential pressure of specification 3.7.A.7.a cannot be maintained, and the differential pressure cannot be restored within the subsequent six (6) hour period, an orderly shutdown shall be initiated and the reactor shall be in the Cold Shutdown condition within the following 24 hours.	

Due to the nitrogen addition, the pressure in the containment after a LOCA could possibly increase with time. Under the worst expected conditions the containment pressure will reach 30 psig in approximately 70 days. If and when that pressure is reached, venting from the containment shall be manually initiated. The venting path will be through the Standby Gas Treatment System in order to minimize the offsite dose.

Following a LOCA, periodic operation of the drywell and torus sprays may be used to assist the natural convection and diffusion mixing of hydrogen and oxygen.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell suppression chamber differential pressure of 1.10 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.33 to 3.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces. Design details are described in References 5 and 6.

7. Standby Gas Treatment System and Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shut down and the drywell is chargers for the 125 volt station batteries, and one of the two 250 volt battery chargers shall be operable.

4. The emergency 4160 volt buses 1A3 and 1A4, and 480 volt buses 1B3, 1B4, 1B9 and 1B20 shall be energized and operable.

b. Once per operating cycle the condition under which the diesel-generator is required will be simulated and a test conducted to demonstrate that it will start and accept the emergency load within the specified time sequence. The diesel-generator shall be operated loaded for a minimum of 5 minutes. An interruption of the diesel-generator will then be simulated to demonstrate that upon subsequent reconnection, it will again accept the emergency load within the specified time sequence. The results shall be logged.

c. The quantity of diesel fuel available shall be logged monthly and after each use of the diesels.

d. Once a month a sample of diesel fuel shall be checked for viscosity, water and sediment. The values for viscosity, water and sediment shall be within the acceptable limits specified in Table 1 of ASTM D975-68 and logged.

e. Each diesel-generator shall be given an annual inspection in accordance with instructions based on the manufacturer's recommendations.

f. A sample test and record shall be made of each oil delivery before it is placed in the storage tank.

2. Unit Batteries

a. Every week the specific gravity, the voltage and temperature of the pilot cell and overall battery voltage shall be measured and logged.

4.8 BASES:

The monthly tests of the diesel-generators are conducted to demonstrate satisfactory system performance and operability. The test of the automatic starting circuits will prove that each diesel will receive all automatic start signals. The loading of each diesel-generator is conducted to demonstrate proper operation at maximum expected emergency loading and at equilibrium operating conditions. Generator experience at other generator stations indicates that the testing frequency is adequate to assure a high reliability of operation should the system be required.

Each diesel-generator has two independent starting air supply systems. One consists of a motor driven air compressor which automatically recharges two air receivers and the other consists of a diesel driven air compressor which is manually operated to recharge a third air receiver. During the monthly check of the diesel-generator, both air start systems will be checked for proper operation.

Following the tests (at least monthly) or other operation of the units, the fuel volume remaining in the diesel oil storage tank will be checked.

At the end of the monthly loads test of the diesel-generator, the fuel oil transfer pump will be operated to refill the day tank and to check the operation of this pump. The day tank level indicator and alarm switches and fuel oil transfer pump control switches will be checked at this time.

The test of the diesels once each operating cycle will be more comprehensive in that it will functionally test the system; i.e., it will check starting and closure of breakers and sequencing of loads. The units will be started by simulation of a loss-of-coolant accident. In addition, a loss of normal power condition will be imposed to simulate a loss of off-site power. The timing sequence will be checked to assure proper loading in the time required. After operating for a minimum of 5 minutes, an interruption of the diesel-generator will be simulated. After a load shed, the subsequent reconnection will be checked to assure that loading of the diesel-generator is again through the load sequencer in the time required. Periodic tests check the capability of the units to start in the required time and to deliver the expected emergency load requirements. Periodic testing of the various components plus a functional test each operating cycle are sufficient to maintain adequate reliability.

Logging the diesel fuel supply after each operation (at least monthly) assures that the minimum fuel supply requirements will be

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 58 TO LICENSE NO. DPR-49

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

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

1.0 INTRODUCTION

Iowa Electric Light and Power Company (the licensee) requested amendments to the Technical Specifications for the Duane Arnold Energy Center (DAEC) by letters dated August 30, 1977 (Reference 1), as supplemented March 18, 1980 (Reference 2), October 26, 1977 (Reference 3), October 22, 1979 (Reference 4), and March 4, 1980 (Reference 5). The amendments are associated with degraded grid voltage protection, reactor protection system instrumentation, the end-of-cycle recirculation pump trip, and suppression chamber downcomers, respectively.

2.0 DEGRADED GRID VOLTAGE

2.1 DISCUSSION

The NRC staff requested (Reference 6) the licensee to assess the susceptibility of the safety-related electrical equipment at the DAEC to a sustained voltage degradation of the offsite source and interaction of the offsite and onsite emergency power systems. The licensee was requested to either propose modifications to satisfy the positions and criteria or furnish an analysis to substantiate that the existing facility design has equivalent capabilities. By Reference 1, as revised by Reference 2, the licensee proposed certain design modifications and changes to the Technical Specifications to satisfy Reference 6.

2.1.1 Current DAEC Design

The current design uses the following scheme for undervoltage protection and load shedding for each of the two 4160V class 1E buses:

(1) One relay on the transformer side of each of the two offsite power supply breakers is set at 65% of 4160V. These relays will trip their respective feeder breaker and provide a fast transfer permissive to the second offsite source. If both offsite sources drop below 65% of nominal, then these relays together will start the corresponding diesel generator and provide a loss of offsite power (loop) signal.

(2) One relay on the bus is set at 65% of 4160V to provide permissive for load sequencing.

(3) One relay on the bus is set at 24%±4% of 4160V to initiate load shedding.

2.1.2 Proposed Modification

The licensee has proposed adding eight definite-time undervoltage relays to the two 4160V class 1E buses. There would be four relays per bus, arranged in a one-out-of-two-twice coincident logic. These relays will trip the respective bus incoming breaker when a voltage below 92.2% of nominal (4160V) persists for 8.5 seconds on either bus.

Proposed changes to the plant's Technical Specifications would add the surveillance requirements, allowable limits for the setpoint and time delay, and limiting conditions for operation for the second-level undervoltage monitors.

2.2 EVALUATION

2.2.1 Second-Level Voltage Protection

Reference 6 required that a second level of undervoltage protection for the onsite power system be provided and stipulated criteria that the undervoltage protection must meet. The criteria pertained to: (1) the selection of voltage and time setpoints, (2) the inclusion of coincidence logic, (3) selection of the time delay, (4) automatic disconnection of offsite power sources, (5) compliance with the requirements of IEEE Standard 279-1971 (Reference 8), and (6) Technical Specifications for the second-level voltage protection monitors.

The proposed setpoint for the second level undervoltage relay is based on the limiting value of voltage for continuous operation of class 1E equipment. This voltage is 86% of 480V and 92.2% of 4160V. The licensee's analysis establishes that all class 1E equipment will be adequately protected from the sustained degraded grid voltage by means of the proposed relays set at 92.2% of 4160V.

The proposed modification incorporates a one-out-of-two-twice coincident logic scheme to preclude spurious trips of the offsite power sources.

The proposed time delay: (1) does not exceed the maximum time delay as analyzed in the DAEC FSAR, (2) will not cause any thermal damage to safety-related equipment, and (3) has a setpoint within voltage ranges recommended by ANSI C84.1-1977 (Reference 7) for sustained operation. The time delay is long enough to override any short duration disturbance that may reduce the offsite power sources and, additionally, will not cause failure of safety-related equipment since the voltage setpoint is within the allowable tolerance of the equipment rated voltages.

The proposed modification automatically initiates the disconnection of offsite power sources whenever the voltage setpoint and time delay limits have been exceeded.

The proposed modification is designed to meet IEEE Std. 279-1971. The logic for each bus is completely independent and separated from the logic for the other bus. Coincidence logic is used so that no single blown P.T. fuse or malfunctioning relay will prevent a safety action when it is required. All of the equipment is seismically qualified. The logic is capable of being tested and calibrated during power operation.

The proposed Technical Specification changes comply with the requirements of Reference 6 except for the frequency of functional tests of the existing undervoltage protection relays. The circuit design is such that these relays can be tested only by providing jumpers across the circuit and thus should be tested only once per operating cycle (during plant shutdown). The licensee has agreed (Reference 2) to modify the design of these relays, to allow monthly functional testing, during the 1981 refueling outage. The licensee will propose appropriate changes in Technical Specifications for our review during the 1981 refueling outage.

The proposed modifications comply with the staff position (Reference 6) concerning second-level voltage protection. All of the staff's requirements and design base criteria have been met. The modifications will protect the class 1E equipment from a sustained degraded voltage condition of the offsite power source.

2.2.2 Load Shed

Reference 6 required that the system design automatically prevent load shedding of the emergency buses once the onsite sources are supplying power to all sequenced loads. The load shedding must also be reinstated if the onsite breakers are tripped. However, if an adequate basis can be provided for retaining the load shed feature when the loads are energized by the onsite power source, then the undervoltage relay setpoint initiating the load shed should be assigned maximum and minimum limits.

The licensee's proposal retains the load shedding feature on the emergency buses when the onsite source is supplying power to these buses. The basis is that the load shedding relays are set at 20.2% of nominal voltage and, during the sequencing of loads on the diesel generators, bus voltage does not drop below 72 percent of nominal. Thus, a load shed will not recur due to motor starting inrush currents. But if a diesel generator breaker trips or a diesel generator voltage drops below 20.2 percent for any reasons, a load shed of the respective bus will recur. Thus, the diesel generator breaker can close again and the sequencing of loads on the diesel generator can begin again when the diesel generator voltage returns. The licensee has assigned maximum and minimum limits to the load shedding relays setpoint.

The basis for retention of the load shedding feature when onsite source is supplying power is adequate and the assignment of maximum and minimum limits to the relays initiating load shedding is in compliance with Reference 6.

2.2.3 Onsite Power Source Testing

Reference 6 required that certain test requirements be added to the Technical Specifications. These tests are to demonstrate the full-functional operability and independence of the onsite power sources and are to be performed at least once per 18 months, during plant shut-down. The tests are to simulate loss of offsite power in conjunction with a simulated safety injection actuation signal and to simulate interruption and subsequent reconnection of onsite power sources. These tests verify the proper operation of the load shed system and that there is no adverse interaction between the onsite and offsite power sources.

The testing procedures proposed by the licensee comply with the full intent of this position. Load shedding on offsite power trip is tested. Load sequencing once the diesel generator is supplying the safety buses is tested. An interruption of the diesel generator and its subsequent reconnection to class 1E buses to accept the emergency loads within the specified time sequence is tested. The time duration of the tests (five minutes with full safety loads) will verify that the time delay is sufficient to avoid spurious trips.

The proposed changes to the Technical Specifications adequately test the system modifications and comply with Reference 6.

Based on the above evaluation, we conclude that the proposed modifications and Technical Specification changes are acceptable.

3.0 REACTOR PROTECTION INSTRUMENTATION

3.1 DISCUSSION

By Reference 3, the licensee requested an amendment to the DAEC Technical Specifications to modify a setpoint for reactor protection system instrumentation. The modification would lower the trip setpoint for initiation of the recirculation pump trip (RPT), associated with an anticipated transient without scram (ATWS), on high reactor pressure from 1135 psig to 1120 psig. The licensee's plant-unique analysis for ATWS-RPT assumed a pressure setpoint of 1135 psig; however, the allowable instrument tolerance of ±15 psig could give a setpoint of 1150 psig, which is non-conservative with respect to the analysis. The licensee determined that the proposed modification would ensure initiation of the ATWS-RPT within the pressure setpoint assumed in the plant-unique analysis when allowable instrument tolerance is considered.

3.2 EVALUATION

The ATWS-RPT system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. Reference 9 addresses the plant response to this postulated event and the pressure setpoint for ATWS-RPT initiation.

The proposed modification is conservative in that, when compared with the current Technical Specifications, the ATWS-RPT would be initiated earlier in the transient scenario and thus tend to further mitigate the consequences of the transient. The staff, therefore, finds the proposed modification acceptable.

4.0 END-OF-CYCLE RECIRCULATION PUMP TRIP

4.1 DISCUSSION

By Reference 4, the licensee applied to amend the DAEC Technical Specifications for installation of the end-of-cycle (EOC) recirculation pump trip (RPT). Reference 10, enclosed to Reference 4, provided the design of the EOC-RPT.

The purpose of the EOC-RPT is to provide significant improvement in the thermal margin of DAEC by reducing the severity of possible pressurization transients. The EOC-RPT accomplishes this purpose by rapidly cutting off power to the recirculation pump motors during generator load rejection (turbine control valve fast closure) or turbine trip (stop valve closure). The result of rapid power interruption is a rapid reduction in recirulation flow and a corresponding increase in the core void content during the pressurization transients, thereby reducing the peak transient power and heat flux. For DAEC, either a generator load rejection or a turbine trip (without bypass valve operation) is the limiting thermal event near the end of each fuel cycle. Since the EOC-RPT provides improved thermal margin for these limiting events, a reduction can be realized in the operating limit minimum critical power ratio.

4.2 EVALUATION

The attached interim report ("Technical Evaluation of the End-of-Cycle Recirculation Pump Trip for Duane Arnold Energy Center") was prepared for use by Lawrence Livermore Laboratories, as part of our technical assistance program. This report provides a technical evaluation of the electrical instrumentation and control design aspects of the EOC-RPT and is based upon the criteria identified in Section 7.1 of Appendix B of the staff's "Safety Evaluation of the Duane Arnold Energy Center" dated January 23, 1973.

In addition, the staff has reviewed the licensee's revision (Reference 11) of the Technical Specifications for the plant to assure that the response time testing recommended by our consultant is included.

Based upon our review of the consultant's evaluation and the plant Technical Specifications, we conclude that the electrical, instrumentation and control design aspects of the EOC-RPT are acceptable. We also consider the contractor's interim report to be a final report because there is to be no further contractor effort on this subject for this plant.

5.0 SUPPRESSION CHAMBER DOWNCOMERS

5.1 DISCUSSION

By Reference 5, the licensee requested an amendment to the DAEC Technical Specifications in conjunction with Mark I Containment Long Term Program (LTP) modifications. The proposed amendment would reduce the minimum suppression chamber downcomers submergence to 3.0 feet and would reduce the minimum differential pressure between the drywell and the suppression chamber from 1.30 to 1.10 psid. The licensee is shortening the length of downcomers as part of the LTP and, additionally, has determined that the existing Technical Specification limitation on drywell to suppression chamber differential pressure cannot be maintained with the shortened downcomers.

2 EVALUATION

One method of suppression pool hydrodynamic load mitigation that the Mark I Owners Group has adopted for the LTP is reducing the initial submergence of the downcomer in the suppression pool to a minimum of three feet. By shortening the length of the downcomer, the pool volume (i.e., thermal capacity) of the original design would be maintained. This approach, however, raises concern regarding the increased potential for uncovering the downcomers and steam condensation capability, both of which could lead to torus overpressurization.

5.2.1 Seismic Slosh

The potential for downcomer uncovery is addressed in the assessment of seismic slosh. This assessment was performed at the most extreme conditions that could potentially lead to uncovering of the downcomers and was predicted on a minimum three-foot downcomer submergence.

Seismic motion induces suppression pool waves which can (1) impart an oscillatory pressure loading on the torus shell, and (2) potentially lead to uncovering the ends of the downcomers, which would result in steam bypass of the suppression pool and potential overpressurization of the torus, should the seismic event occur in conjunction with a Loss of Coolant Accident (LOCA). To assess these effects, the Mark I Owners Group undertook the development of an analytical model which would provide plant-specific seismic wave amplitudes and torus wall pressures. This model was based on 1/30-scale "shake test" data for a Mark I torus geometry (Reference 12).

Based on the results of plant-specific analyses, using the analytical model, the Mark I Owners Group concluded that (1) the seismic wave pressure loads on any Mark I torus are insignificant in comparison with the other suppression pool dynamic loads, and (2) the seismic wave amplitudes will not lead to uncovering the downcomers for any Mark I plant. This conclusion was based on the maximum calculated pressure loads and the minimum wave through depth relative to the downcomer exit.

We have reviewed comparisons of the analytical predictions with scaled-up test data, the small-scale test program, and the seismic spectrum envelope used in the plant-specific analyses. Based on this review, we conclude that the seismic slosh analytical predictions will provide reasonably conservative estimates of both the wall pressure loading and the wave amplitude, for the range of Mark I plant conditions.

Since the maximum local wall pressures were found to be less than 0.8 psi at a 95% upper confidence limit, the Mark I Owners Group has proposed that the seismic slosh loads may be neglected in the structural analysis. We agree that the seismic slosh loads are insignificant in comparison with the other suppression pool dynamic loads. On this basis, we conclude that neglecting seismic slosh loads for the plant-unique analyses is acceptable.

The results of the slosh wave amplitude predictions indicate that, within the local area of maximum amplitude and with maximum suppression pool drawdown (resulting from ECCS system flows), the slosh waves will not cause uncovering of the downcomers. We have reviewed the assumptions used in these analyses and conclude that they are sufficiently conservative. Based on the above discussion, we find the proposed change acceptable.

5.2.2 Condensation Capability

Condensation capability of the suppression pool is a function of the local pool temperature in the vicinity of the downcomer exit. Full Scale Test Facility (FSTF) test results (Reference 13) and foreign test data (Reference 14) have shown that thermal stratification occurs, and becomes more severe as the downcomer submergence is reduced. The most severe thermal stratification has been observed in low flow tests with a quiescent pool. However, in actual plant conditions, the Residual Heat Removal (RHR) system and Safety Relief Valve (SRV) discharge provide sufficient long-term pool mixing to minimize thermal stratification. Even with verticle thermal stratification, we have determined that the high energy reposition is accompanied by an increased flow and mixing, which prevent overpressurization of the torus. In addition, the analytical predictions of the torus pressure and bulk temperature response have been found to be conservative when compared with FSTF test data for plant simulated initial conditions. The local temperature variation in the pool which has been observed in the test data is not significant to the structure, and, therefore, need not be considered in the structural analysis.

Based on this assessment, we conclude that a minimum initial downcomer submergence of three feet is acceptable, and there is sufficient conservatism in the containment response analysis techniques to accommodate the effects of thermal stratification.

5.2.3 Differential Pressure

The introduction of a positive pressure differential between the drywell and the suppression chamber air volume reduces the height of the water leg inside the downcomers. The reduced water leg permits the downcomers to clear earlier in the LOCA transient with the drywell consequently at a lower pressure. This effect reduces both the downward and upward pressure loads on the torus. The DAEC plant-unique minimum differential pressure was reviewed and approved (Reference 15) by the staff as part of the Short Term Program (STP).

To evaluate the proposed modification to the previously approved minimum differential pressure, the staff evaluated the combined effects of both downcomer shortening and the reduction in differential pressure on torus pressure loading. A sensitivity analysis was performed based upon the methodology accepted by Section C of Reference 15. The effect of the downcomer shortening taken independently, was a reduction in both the downward and upward torus pressure loads. The effect of reducing differential pressure taken independently, was an increase in both downward and upward torus pressure loads. The staff found the net effect, however, to be a reduction in the magnitude of both the downward and the upward torus pressure loads as compared with those found acceptable by Reference 15.

Since the proposed modification provides an increased safety margin for torus pressure loads previously found acceptable for the STP (Reference 15), we conclude that, in the interim until the LTP is completed, the proposed modification is acceptable. Therefore, we find the proposed Technical Specifications acceptable.

6.0 ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 10, 1980

Attachment:
Technical Evaluation of the
End-of-Cycle Recirculation
Pump Trip for the Duane
Arnold Energy Center

References

1. Letter, L. Liu (Iowa Electric) to G. Lear (NRC), dated August 30, 1977.

2. Letter, L. D. Root (Iowa Electric) to H. Denton (NRC), dated March 18, 1980.

3. Letter, L. Liu (Iowa Electric) to E. Case (NRC) dated October 26, 1977.

4. Letter, L. D. Root (Iowa Electric) to H. Denton (NRC), dated October 22, 1979.

5. Letter, L. D. Root (Iowa Electric) to H. Denton (NRC) dated March 4, 1980.

6. Letter, G. Lear (NRC) to D. Arnold (Iowa Electric) dated June 2, 1977.

7. ANSI C84.1-1977, "Voltage Ratings for Electric Power Systems and Equipment" (60HZ)

8. IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

9. Final Safety Analysis Report, Duane Arnold Energy Center.

10. General Electric Company, "Basis for Installation of Recirculation Pump Trip System - Duane Arnold Energy Center", NEDO-24220, September 1979.

11. Letter, L. D. Root (Iowa Electric) to H. Denton (NRC), dated April 9, 1980.

12. S. M. Arian, "Mark I Containment Program Seismic Slosh Evaluation" GE Proprietary Report NEDE-023702-P, March 1978.

13. G. W. Fitzsimmons and others, "Mark I Containment Program Full Scale Test Program Final Report" GE Proprietary Report NEDE-2453q-P, April 1979.

14. K. W. Wong, "Mark I Containment Program Downcomer Reduced Submergence Functional Assessment Report" General Electric Proprietary Report NEDE-21885-P, June 1978.

15. NRC Report, "Mark I Containment Short Term Program - Safety Evaluation Report," NUREG-0408, December 1977.

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TECHNICAL EVALUATION OF THE END-OF-CYCLE RECIRCULATION PUMP TRIP FOR THE DUANE ARNOLD ENERGY CENTER

(Docket No. 50-331)

L. R. Peterson

March 1980

Lawrence Livermore Laboratory

This is an informal report intended primarily for internal or limited external distribution. The opinions and conclusions stated are those of the author and may or may not be those of the Laboratory.

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ABSTRACT

This report documents the technical evaluation of the end-of-cycle recirculation pump trip for the Duane Arnold Energy Center. The review criteria are based on IEEE Std-279-1971, IEEE Std-323-1974, IEEE Std-338-1977, and General Design Criteria 13, 20 through 24, and 29 of the Code of Federal Regulations, Title 10, Part 50, Appendix A requirements for determining the acceptability of the proposed system.

FOREWORD

This report is supplied as part of the Selected Electrical, Instrumentation, and Control Systems Issues (SEICSI) Program being conducted for the U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Operating Reactors, by Lawrence Livermore Laboratory, Engineering Research Division of the Electronics Engineering Department.

The U. S. Nuclear Regulatory Commission funded the work under the authorization entitled "Electrical, Instrumentation and Control System Support," B&R 20 19 04 031, FIN A-0231.

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TECHNICAL EVALUATION OF THE END-OF-CYCLE RECIRCULATION PUMP TRIP FOR DUANE ARNOLD ENERGY CENTER

(Docket No. 50-331)

L. R. Peterson
Lawrence Livermore Laboratory, Nevada

1. INTRODUCTION

Iowa Electric Light and Power Company (IEL&P) by its letter dated October 22, 1979 [Ref. 1], applied to amend the Technical Specifications for the Duane Arnold Energy Center (DAEC) for installation of end-of-cycle (EOC) recirculation pump trip (RPT). The design of the EOC-RPT feature for DAEC is described in General Electric Company (GE) report NEDO-24220, "Basis for Installation of Recirculation Pump Trip System - Duane Arnold Energy Center," [Ref. 2], which was submitted with the proposed change to the DAEC Technical Specifications, RTS-116 [Ref. 3].

The EOC-RPT feature is installed to improve the thermal margin of a boiling water reactor (BWR) near the end of each fuel cycle by reducing the severity of possible pressurization transients. The RPT accomplishes this objective by rapidly cutting off power to the recirculation pump motors during generator load rejection (turbine control valve fast closure) or turbine trip (stop valve closure). This results in a rapid reduction in recirculation flow and increases the core void content during a pressurization transient, thereby reducing the peak transient power and heat flux.

2. DESIGN DESCRIPTION

The design for the EOC-RPT installation at the Duane Arnold Energy Center is described in a GE report, NEDO-24220 [Ref. 2]. The EOC-RPT is part of the reactor protection system (RPS) because it is an essential supplement to the reactor scram system. All components of the EOC-RPT system are Class 1E.

The EOC-RPT is required to quickly shut down both BWR coolant recirculation pumps when closure of all four turbine stop valves occurs, or when fast closure of all four turbine control valves occurs. An EOC-RPT trip may occur, but is not required, when one turbine stop valve or one turbine control valve remains open. To mitigate pressurization transient effects, the EOC-RPT must shut down the recirculation pumps within approximately 200 ms after initial closure movement of either the turbine stop valves or the turbine control valves.

The EOC-RPT installation is composed of sensors that detect closure of the turbine stop valves or fast closure of the turbine control valves ~~and~~ combined with relays, logic circuits, and fast-acting circuit breakers that interrupt the current from the recirculation pump motor-generator set generators to the recirculation pump motors. When the redundant RPT breakers trip open, the recirculation pumps coast down under their own inertia. To satisfy the reactor protection system (RPS) single-failure criterion, the EOC-RPT has two almost identical divisions that actuate RPT in a one-out-of-two configuration. Either of the two RPT divisions operates independent breakers in the supply circuits of both recirculation pumps.

Turbine stop valve closure is detected by four position switches that open when the associated stop valves are less than 90 percent open. Turbine control valve fast closure is detected by four pressure switches in the hydraulic control system for the valves. The pressure switches open when the hydraulic control fluid pressure decreases below the trip level. The stop valve position sensors and the control valve hydraulic pressure sensors for RPT are the same ones used in the reactor scram system to initiate scram when turbine stop valve closure or turbine control valve fast closure occurs.

The actuation of any RPT sensor causes an associated electromagnetic relay to de-energize. The contacts of these relays are combined in logic circuits with contacts from an operating bypass and contacts from a key-controlled manual bypass switch. The logic circuits control current to the trip circuits of the RPT circuit breakers. The operating bypass disables the RPT system when turbine first-stage pressure is less than that for 30 percent reactor power. The same operating bypass concurrently disables the turbine inputs to the scram system. A manual bypass switch allows each RPT division to be disabled and placed out of service for maintenance or testing.

The fast-closure sensors from each of two turbine control valves provide inputs to one RPT division and the sensors from the other two turbine control valves provide inputs to the second RPT division. Similarly, the position switches from each of two turbine stop valves provide inputs to one RPT division and position switches from the other two stop valves provide inputs to the other RPT division. The sensor relay contacts for each RPT division are arranged to form a two-out-of-two logic for the fast closure of control valves and a two-out-of-two logic for closure of the stop valves. The operation of either logic in a RPT division will actuate the EOC-RPT feature.

3. EVALUATION

The EOC-RPT feature is part of the reactor protection system and is an essential supplement to the reactor scram function. The EOC-RPT is required to comply with the criteria of IEEE Std-279-1971 [Ref. 4], IEEE Std-323-1974 [Ref. 5], and IEEE Std-338-1977 [Ref. 6] and with General Design Criteria 13, 20 through 24, and 29 of 10 CFR 50, Appendix A [Ref. 7].

The EOC-RPT system at Duane Arnold Energy Center is similar to that previously approved by NRC for Browns Ferry, Unit 1. The two RPT divisions are physically and electrically independent. The sensors and relays providing inputs to the RPT systems originate from two separate Class 1E scram channels. The signal channels are properly grouped and separated to provide independence between the corresponding scram channels and the associated RPT divisions. The scram and RPT logic relays are fail-safe and will go to the tripped state on loss-of-power or loss-of-input signal from each sensor.

The RPT circuit breaker control and trip circuits are not fail-safe, and will not trip on loss of power. The RPT circuit breakers that interrupt the current to the recirculation pumps require power to actuate. For this reason, the RPT logic circuits, control circuits, and trip circuits operate on 125 Vdc. Each RPT division is supplied by a separate Class 1E-rated 125 Vdc battery power supply with 30 amp inline fuses for the positive and negative lines from the battery supply.

Separate 10 amp branch fuses protect the RPT circuit breaker elevating-mechanism circuits; 15 amp branch fuses protect the local breaker closing circuits, the control room breaker closing circuits, and the indicator light circuits. These branch fuses isolate the circuits from the breaker trip circuits so that a short circuit in the elevating or closing functions of the breaker will not disable breaker trip actuation. A relay in each RPT division senses loss of power to the trip circuit in that division and actuates an "RPT Power Not Available" annunciator in the control room. In addition, indicating lights are provided in the control room to monitor the trip coil circuits and the position of the trip breakers. The NRC has previously found that this departure from fail-safe design is acceptable.

There is one interconnection between each EOC-RPT division and a non-safety system. When each RPT breaker trips, auxiliary relay contacts in the RPT breaker actuate a control circuit for the recirculation pump motor-generator (M-G) set to de-energize the M-G set after the RPT breaker interrupts the current from the M-G set to the recirculation pump motor. This interlock is adequately isolated so that no credible failure can prevent proper RPT action.

An operating bypass automatically disables the RPT system when the reactor is operating at less than 30 percent power. The operating bypass is annunciated automatically in the control room.

Each RPT division can be bypassed manually by use of an out-of-service keyswitch which is administratively controlled. Use of the out-of-service keyswitch bypass produces a suitable annunciator indication in the control room when the keyswitch is turned to the "RPT SYS INOP" position.

The proposed technical specifications for the Duane Arnold Energy Center provide suitable restrictions to limit operating power when one or both of the EOC-RPT divisions are inoperable.

Capability to check the RPT sensors and logic is provided by operating each valve, one at a time. Lights across the relay contacts in the logic indicate proper operation at that point. The RPT divisions do not need to be bypassed to conduct such tests. During periodic testing of the scram logic, when two turbine stop valves are operated simultaneously, the affected RPT division must be bypassed briefly to prevent RPT actuation. The bypass is accomplished by use of the EOC-RPT system out-of-service keyswitch during the scram-logic test.

The proposed technical specifications for Duane Arnold Energy Center specify monthly functional checks of both the EOC-RPT initiate logic and scram logic. We consider monthly testing of the EOC-RPT input sensors and logic circuits to be adequate for providing timely indications of system failure.

Although the purpose of the RPT is to mitigate a core-wide pressurization transient, the desired thermal margin advantage can be realized only if the initiating events are sensed on an anticipatory basis, rather than by monitoring reactor pressure directly. The use of pressure switches to sense the loss of hydraulic control fluid pressure to each turbine control valve is adequate to anticipate fast closure of those valves. Similarly, position switches set to trip at 90 percent open will adequately anticipate closure of the turbine stop valves. The EOC-RPT is not given credit for any other initiating events.

To be effective, the RPT must be initiated almost immediately. GE states that their analysis shows that manual initiation of a prompt trip of the recirculation pumps, at any reasonable point after the time when automatic action should have occurred, will not produce a significant improvement on the situation. The power to the recirculation pump motor-generator sets can be tripped manually from the control room. Therefore, provisions for manual initiation of the EOC-RPT feature are unnecessary.

The RPT feature is required to reduce recirculation-pump flow after either the turbine control valves or the stop valves start closing, and within a delay time assumed in the transient calculations for that operating cycle. The licensee and GE have specified that the RPT circuit breakers will have a maximum interrupting time of 135 ms. The remainder of the shutdown time will include system action, sensor response, logic response, and pump coastdown.

The licensee and GE include time-response tests during initial testing of the EOC-RPT installation to confirm that the system time response, consisting of coastdown time and delay time, is less than that assumed in the applicable transient calculations. However, the licensee has not included any provisions in the technical specifications for subsequent time-response testing that would detect any degradation of system time response.

We concur with the proposed surveillance requirement in the Duane Arnold Energy Center Technical Specifications that functional tests of the EOC-RPT circuit breakers be conducted once per operating cycle. We further recommend testing the EOC-RPT system response time from initial closure movement of the turbine stop valves or control valves until recirculation pump shutdown or, alternatively, measuring the RPT circuit breaker interrupting time with suitable correlation of that measurement to the EOC-RPT system response time. To better meet the criteria of IEEE Std-338-1977, Section 6.3.4, these time-response tests should be made prior to each operating cycle for both EOC-RPT divisions as part of the RPT circuit breaker functional checks.

4. CONCLUSIONS

Considering the separation, independence, and isolation of the two EOC-RPT divisions and their respective inputs, circuits, and power supplies, the EOC-RPT feature for the Duane Arnold Energy Center meets the criteria of IEEE Std-279-1971, IEEE Std-323-1974 and General Design Criteria 13, 20 through 24, and 29 of 10 CFR 50, Appendix A. We recommend approval of the EOC-RPT system design as submitted by the licensee. We also recommend approval of the proposed change for the addition of an EOC-RPT feature at Duane Arnold Energy Center to DAEC Technical Specifications (RTS-116).

To better fulfill the criteria of IEEE Std-338-1977, we recommend that suitable requirements for time-response tests of the EOC-RPT system prior to each operating cycle be included in the Surveillance Requirements of the Technical Specifications. Such time-response tests should be designed to verify that the EOC-RPT system response time is less than the response time assumed in the applicable transient calculations for the corresponding operating cycle.

REFERENCES

1. Iowa Electric Light and Power Company letter (L. D. Root) to NRC/NRR (H. Denton) dated October 22, 1979.

2. General Electric Company, Nuclear Power Systems Division, "Basis for Installation of Recirculation Pump Trip System - Duane Arnold Energy Center," report NEDO-24220, 79 NED306, Class I, September 1979.

3. Iowa Electric Light and Power Company, "Proposed Change RTS-116 to DAEC Technical Specifications," enclosure to IEL&P letter dated October 22, 1979.

4. IEEE Std-279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

5. IEEE Std-323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

6. IEEE Std-338-1977, "Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems."

7. Code of Federal Regulations, Title 10, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants."

ATTACHMENT TO LICENSE AMENDMENT NO. 58

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 pages are identified by Amendment Number and contain vertical lines indicating the area of change.

3.2-4

3.2-14

3.2-15

3.2-23

3.2-26

3.2-34

3.7-14

3.7-41

3.8-2

*3.8-11

3.8-12

*No change. Provided for convenience

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-331

IOWA ELECTRIC LIGHT AND POWER COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 58 to Facility Operating License No. DPR-49 issued to Iowa Electric Light and Power Company, Central Iowa Power Cooperative, and Corn Belt Power Cooperative, which revises the Technical Specifications for operation of the Duane Arnold Energy Center, located in Linn County, Iowa. The amendment is effective as of the date of its issuance.

The amendment incorporates provisions into the Technical Specifications for (1) modifications associated with degraded grid voltage protection, (2) installation of the end-of-cycle recirculation pump trip, (3) modification of a reactor protection instrumentation set point, and (4) modifications associated with the containment suppression chamber.

The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated August 30, 1977 (supplemented March 18, 1980), October 22, 1979 (supplemented April 9, 1980), and March 4, 1980, (2) Amendment No. 58 to License No. DPR-49, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 10th day of April 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

T A Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors