

August 9, 1990

Docket Nos. 50-237  
and 50-249

Mr. Thomas J. Kovach  
Nuclear Licensing Manager  
Commonwealth Edison Company-Suite 300  
OPUS West III  
1400 OPUS Place  
Downers Grove, Illinois 60515

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<b>Docket Files</b>	NRC & Local PDRs
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JCalvo	ACRS(10)
GPA/PA	OC/LFMB
PDIII-3 Gray	CMcCracken
Crutchfield	

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NOS. 77193 AND 77194)

The Commission has issued the enclosed Amendment No. 112 to Provisional Operating License No. DPR-19 for Dresden Unit 2 and Amendment No. 108 to Facility Operating License No. DPR-25 for Dresden Unit 3.

The amendments exclude the Reactor Building Closed Cooling Water inlet pathway to the primary containment for each unit from Appendix J, Type C leak rate testing requirements until the next refueling outage for each unit. A temporary waiver of compliance was approved orally on July 20, 1990 and in writing on July 24, 1990. The waiver allowed the staff sufficient time to process your emergency technical specification amendment submittal of July 31, 1990.

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

Sincerely,

Original Signed By:

Byron L. Siegel, Project Manager  
Project Directorate III-2  
Division of Reactor Projects - III,  
IV, V and Special Projects  
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 112 to License No. DPR-19
2. Amendment No. 108 to License No. DPR-25
3. Safety Evaluation

cc w/enclosures:  
See next page

9008160039 900809  
PDR ADOCK 05000237  
P PIC

DOCUMENT NAME: AMENDMENT 77193/94

Office:	LA/PDIII-2	PM/PDIII-2	PD/PDIII-2	BC/SPLV
Surname:	CMoore	BSiegel	RBarrett	CMcCracken
Date:	08/7/90	08/7/90	08/19/90	08/10/90

*Concurred verbally 8/10/90*

*J. Pol III*

*ET Holler*

*08/8/90*

*SUBJECT TO SAFETY COMMITMENTS RE: EVALUA.*



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

August 9, 1990

Docket Nos. 50-237  
and 50-249

Mr. Thomas J. Kovach  
Nuclear Licensing Manager  
Commonwealth Edison Company-Suite 300  
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Dear Mr. Kovach:

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The amendments exclude the Reactor Building Closed Cooling Water inlet pathway to the primary containment for each unit from Appendix J, Type C leak rate testing requirements until the next refueling outage for each unit. A temporary waiver of compliance was approved orally on July 20, 1990 and in writing on July 24, 1990. The waiver allowed the staff sufficient time to process your emergency technical specification amendment submittal of July 31, 1990.

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

Sincerely,

A handwritten signature in cursive script that reads "Byron L. Siegel".

Byron L. Siegel, Project Manager  
Project Directorate III-2  
Division of Reactor Projects - III,  
IV, V and Special Projects  
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 112 to License No. DPR-19
2. Amendment No. 108 to License No. DPR-25
3. Safety Evaluation

cc w/enclosures:  
See next page

Mr. Thomas J. Kovach  
Commonwealth Edison Company

Dresden Nuclear Power Station  
Unit Nos. 2 and 3

cc:

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Morris, Illinois 60450

U. S. Nuclear Regulatory Commission  
Resident Inspectors Office  
Dresden Station  
Rural Route #1  
Morris, Illinois 60450

Chairman  
Board of Supervisors of  
Grundy County  
Grundy County Courthouse  
Morris, Illinois 60450

Regional Administrator  
Nuclear Regulatory Commission, Region III  
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Glen Ellyn, Illinois 60137

Illinois Department of Nuclear Safety  
Office of Nuclear Facility Safety  
1035 Outer Park Drive  
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State of Illinois Center  
100 W. Randolph  
Suite 11-300  
Chicago, Illinois 60601



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 112  
License No. DPR-19

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by the Commonwealth Edison Company (the licensee) dated July 31, 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 3.B. of Provisional Operating License No. DPR-19 is hereby amended to read as follows:

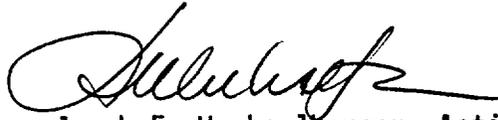
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P PDC

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 112, 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



Jacob F. Wechselberger, Acting Director  
Project Directorate III-2  
Division of Reactor Projects - III,  
IV, V and Special Projects  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: August 9, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 112

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE

3/4.7-6

3/4.7-7

3/4.7-8

3/4.7-9

INSERT

3/4.7-6

3/4.7-7

3/4.7-8

3/4.7-9

3/4.7-9a

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

- (1) An overall integrated leakage rate for Type A tests of:
  - (a)  $L_{am}$  less than or equal to 75 percent of  $L_a$ . The RBCCW pathway described in Section 4.7.A.2.e.(4) is excluded until the end of the Cycle 12 refueling outage.
  - (b)  $L_{tm}$  less than or equal to 75 percent of  $L_t$ .
- (2) (a) A combined leakage rate of less than or equal to 60 percent of  $L_a$  for all testable penetrations and isolation valves subject to Type B and C tests except for main steam isolation valves. The RBCCW pathway described in Section 4.7.A.2.e.(4) is also excluded until the end of the Cycle 12 refueling outage.

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

(b) A leakage rate of less than or equal to 3.75 percent of  $L_a$  for any one air lock when pressurized to 10 psig.

(c) 11.5 SCF per hour for any main steam isolation valve at a test pressure of 25 psig.

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

c. If two consecutive Type A tests fail to meet either 75 percent of  $L_a$  or 75 percent of  $L_t$ , a Type A test shall be performed at each shutdown for refueling or approximately every 18 months until two consecutive Type A tests meet the above requirements, at which time the normal test schedule may be resumed.

d. The accuracy of each Type A test shall be verified by a supplemental test which:

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

- (1) Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 25 percent of  $L_a$  or 25 percent of  $L_t$ .
  - (2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.
  - (3) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 9 scfm.
- e. Type B and C tests shall be conducted at  $P_a$ , at intervals no greater than 24 months except for tests involving:

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

- (1) Main steam line isolation valves which shall be tested at a pressure of 25 psig each operating cycle.
- (2) Bolted double-gasketed seals which shall be tested at a pressure of 48 psig whenever the seal is closed after being opened and each operating cycle.
- (3) Air locks which shall be tested at 10 psig each operating cycle.
- (4) The Reactor Building Closed Cooling Water (RBCCW) inlet pathway to the primary containment (penetration X-123) will not be tested until the end of the Cycle 12 refueling outage.

f. Continuous Leak Rate Monitor

- (1) When the primary containment is inerted, the containment

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

shall be continuously monitored for gross leakage by review of the inerting system make-up requirements.

(2) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

g. The interior surfaces of the drywell shall be visually inspected each operating cycle for evidence of deterioration.



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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 108  
License No. DPR-25

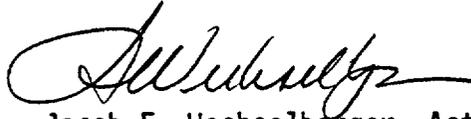
1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by the Commonwealth Edison Company (the licensee) dated July 31, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules 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 3.B. of Facility Operating License No. DPR-25 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 108, 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



Jacob F. Wechselberger, Acting Director  
Project Directorate III-2  
Division of Reactor Projects - III,  
IV, V and Special Projects  
Office of Nuclear Reactor Regulation

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: August 9, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 108

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE

3/4.7-6

3/4.7-7

3/4.7-8

3/4.7-9

INSERT

3/4.7-6

3/4.7-7

3/4.7-8

3/4.7-9

3/4.7-9a

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

(1) An overall integrated leakage rate for Type A tests of:

(a)  $L_{am}$  less than or equal to 75 percent of  $L_a$ . The RBCCW pathway described in Section 4.7.A.2.e(4) is excluded until the end of the Cycle 12 refueling outage.

(b)  $L_{tm}$  less than or equal to 75 percent of  $L_t$ .

(2) (a) A combined leakage rate of less than or equal to 60 percent of  $L_a$  for all testable penetrations and isolation valves subject to Type B and C tests except for main steam isolation valves. The RBCCW pathway described in Section 4.7.A.2.e.(4) is also excluded until the end of Cycle 12 refueling outage.

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

- (b) A leakage rate of less than or equal to 3.75 percent of  $L_a$  for any one air lock when pressurized to 10 psig.
- (c) 11.5 SCF per hour for any main steam isolation valve at a test pressure of 25 psig.

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

- c. If two consecutive Type A tests fail to meet either 75 percent of  $L_a$  or 75 percent of  $L_t$ , a Type A test shall be performed at each shutdown for refueling or approximately every 18 months until two consecutive Type A tests meet the above requirements, at which time the normal test schedule may be resumed.
- d. The accuracy of each Type A test shall be verified by a supplemental test which:
  - (1) Confirms the accuracy of the test by verifying that the difference

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

between the supplemental data and the Type A test data is within 25 percent of  $L_a$  or 25 percent of  $L_t$ .

(2) Has a duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

(3) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 9 scfm.

e. Type B and C tests shall be conducted at  $P_a$ , at intervals no greater than 24 months except for tests involving:

(1) Main steam line isolation valves which shall be tested at a pressure of 25 psig each operating cycle.

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

(2) Bolted double-gasketed seals which shall be tested at a pressure of 48 psig whenever the seal is closed after being opened and each operating cycle.

(3) Air locks which shall be tested at 10 psig each operating cycle.

(4) The reactor Building Closed Cooling Water (RBCCW) inlet pathway to the primary containment (penetration X-123) will not be tested until the end of the Cycle 12 refueling outage.

f. Continuous Leak Rate Monitor

(1) When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system make-up requirements.

3.7 LIMITING CONDITION FOR OPERATION  
(Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS  
(Cont'd.)

(2) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

g. The interior surfaces of the drywell shall be visually inspected each operating cycle for evidence of deterioration.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
SUPPORTING AMENDMENT NO. 112 TO PROVISIONAL OPERATING LICENSE NO. DPR-19  
AND AMENDMENT NO. 108 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated July 31, 1990, Commonwealth Edison Company (CECo) submitted emergency Technical Specification (TS) changes for the Dresden Nuclear Power Station, Units 2 and 3. The proposed change excludes the Reactor Building Closed Cooling Water (RBCCW) system inlet pathway to the primary containment for each unit (two valves) from Appendix J, Type C, leak rate testing requirements until the next refueling outage for each unit (approximately 2 months of continued operation for Dresden, Unit 2 and 10 months of continued operation for Dresden, Unit 3). In addition to not being Type C tested, the two valves in the RBCCW inlet pathway for each unit were not properly included as part of the tested containment boundary during the last containment integrated leak rate test (Type A test). Thus, neither the sum of local leak rates (Type B and Type C) nor the overall integrated containment leak rate (Type A) is known. Because the valves cannot be tested with the plant at power, nor can the lines be isolated by other barriers with the plant at power, the requested changes to the Technical Specifications were needed to allow continued plant operation.

2.0 DISCUSSION AND EVALUATION

The existing RBCCW system has design features that function to inhibit potential release paths. The RBCCW system consists of two 6 inch lines that penetrate primary containment. The supply (inlet) line is normally isolated using a check valve inside and a motor-operated gate valve outside of containment. The return line contains two remotely operated valves, one inside and one outside of the drywell.

In addition to the two containment isolation valves on each line, other barriers exist. Inside of the containment, the piping forms a closed loop. Outside of containment, the piping is configured such that loop water seals are created. The system is filled with water during normal operation. The water serves as a seal in the event the valves leak. Additionally, any through-wall water leaks would be easily detected either inside or outside of the drywell through sump level alarms, system pressures, or tank levels.

The piping outside of containment is connected to a vented surge tank. This tank receives makeup water that is supplied by multiple pumps connected to a common header, which provides suction from a 100,000 gallon storage tank. This configuration provides substantial assurance that the system would remain water-filled in post-accident conditions. Containment leakage, then, would have to enter the closed system inside containment, pass through two valves and a loop seal, and the system would have to be depressurized, before leakage could enter the environment.

In addition to the above, the licensee performed a probabilistic risk assessment (PRA) to further demonstrate that the probability of an event resulting in a loss of containment function coincident with a LOCA during the remainder of the current Dresden (Unit 2 and 3) operating cycles is insignificant. Based on this evaluation, CECO concluded that the fission product barriers remain intact unless an extreme combination of highly improbable coincident failures occurred. The probabilities calculated for the event in which containment function failure would occur under LOCA conditions were found to be less than  $1 \times 10^{-7}$ . A recirculation piping failure (or any high energy line break), RBCCW pipe failure inside containment, and a failure of the loop seal would have to occur in order to result in the failure of the containment function. The probability of a failure of the RBCCW system's containment function coincident with a LOCA was determined to be  $1 \times 10^{-9}$  and is therefore considered to be insignificant.

CECO has also implemented administrative controls that direct the operators to close the remotely operated valves on the RBCCW systems when the Reactor Recirculation Pumps trip during a postulated Loss of Coolant Accident (LOCA). The RBCCW pumps will be kept on, if possible, to ensure the system is filled with water and pressurized above containment pressure. If the RBCCW expansion tank HI/LO level alarm is received during a LOCA event, field teams will be sent, as conditions permit, to check RBCCW piping outside containment to ensure integrity. The Station Director will be informed to take the necessary actions to further isolate the system.

The staff has evaluated the licensee's submittal and concludes, based on the system design, the administrative controls that have been implemented and the low probability of the combination of failures that would have to occur to breach containment integrity, that operation until the next refueling outage with the two valves in the inlet pathway of the RBCCW of each unit untested will not add significantly to the risk to the public health and safety and therefore, the proposed TS changes are acceptable. In addition, a similar TS change was issued to Quad Cities Unit 1 on June 27, 1990.

### 3.0 FINDINGS OF EMERGENCY WARRANTING AN AMENDMENT WITHOUT NOTICE

In the fourth quarter of 1989, the licensee assessed the leak rate testing program at Quad Cities and identified 29 containment pathways that were not being local leak rate tested. The licensee determined the Appendix J, Type C testing was not required for these 29 pathways. However, the NRC staff questioned this determination and on May 18, 1990, concluded that these 29

pathways were, in fact, subject to Appendix J, Type C testing. Once informed of the NRC's position on these pathways, the licensee requested a temporary waiver of compliance. The NRC staff orally granted this waiver on May 18, 1990.

Upon completion of the Quad Cities review in December 1989, these pathways were reviewed against the Dresden local leak rate test (LLRT) program. The RBCCW system was one of the systems identified at Quad Cities, however, it was not identified as a Dresden concern in the December 1989 comparison review with Quad Cities because the RBCCW system had been included in the Dresden LLRT programs since 1984. Subsequent to the December 1989 review, the Corporate General Office Support Services Staff proceeded to conduct a detailed review of the entire Dresden LLRT program and identified five additional pathways of potential concern which were communicated to Dresden Station on July 18, 1990. These pathways were validated by station personnel after further review and walkdowns on July 20, 1990 and a waiver of compliance was orally approved by the staff on the same date. (Three of the pathways identified were leak rate tested within 48 hours and did not require an emergency TS amendment). CECO on July 31, 1990 proposed a TS amendment which deferred the testing of the two remaining RBCCW inlet pathways (one for each unit) until the next refueling outage for each unit.

The staff finds that CECO acted as quickly as possible once informed of the requirement to test these pathways. Furthermore, the staff finds that failure to grant the proposed changes in a timely manner would have required a shutdown of both Dresden units. Accordingly, the staff concludes that the licensee has satisfied the requirements of 10 CFR 50.91(a)(5), and that a valid emergency exists.

#### 4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The NRC staff has reviewed the CECO's amendment application and determined, in accordance with the criteria of 50.92(c), that operation of Dresden, Units 2 and 3, according to the proposed amendment will not:

- (1) Involve a significant increase in the probability or consequences of an accident previously evaluated. Leakage through the associated valves does not alter the initiating aspects of the events. The continued operation in the existing RBCCW configuration does not present a significant increase in the probability of a larger release of radioactivity than described in the FSAR. The existing RBCCW system design features (i.e. water filled system) inhibit potential release paths. In addition, the PRA performed by CECO demonstrated that the probability of containment function failure coincident with a LOCA is not significant (less than  $1 \times 10^{-7}$ ). CECO has also implemented administrative controls to direct operators to close remotely operated valves on the RBCCW systems when a reactor recirculation pump trip occurs during a postulated LOCA. On this basis, it is concluded that continued operation will not result in a significant increase in risk with regards to accident probabilities or consequences.

- (2) Create the possibility of a new or different kind of accident from any accident previously evaluated. The emergency TS amendment does not result in any physical plant changes during the period of interest. If the valves in the RBCCW system inlet pathway should leak, the severity of the accident would at worst be affected, but not the type of accident.
- (3) Involve a significant reduction in the margin of safety. Dose calculations described in the TS Bases suggest that the accident leak rate could be allowed to increase to about 3.2%/day before the guideline thyroid dose value given in 10 CFR 100 would be exceeded. However, 1.6%/day provides an additional margin of safety to assure the health and safety of the general public. Additional margin is further achieved by establishing the allowable operational leak rate at 75% of the maximum allowable leak rate. Despite the lack of rigorous leak testing, substantial barriers to fission product release are provided by the intact system piping and associated valves. These barriers provide mitigating capability such that the potential impact on the margin of safety is insignificant. Additionally, the PRA provided by CECO further demonstrates that the probability of containment function failure coincident with LOCA conditions is also acceptably small.

Accordingly, the Commission has determined that this request does not involve a significant hazards consideration.

#### 5.0 STATE CONSULTATION

The State of Illinois was informed by telephone on August 2, 1990, of the staff's final no significant hazards consideration determination and intent to issue a license amendment. The state contact had no comment.

#### 6.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes 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. The staff has determined that the amendments involve 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 determined that these amendments involve no significant hazards consideration. Accordingly, these amendments meet 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 nor environmental assessment need be prepared in connection with the issuance of these amendments.

7.0 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 (3) the issuance of these amendments will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: Byron Siegel

Dated: August 9, 1990