August 31, 1990

Docket Nos. STN 50-454
STN 50-455
STN 50-456
and STN 50-457

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

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. 76715, 76716, 76717 AND 76718)

The Commission has issued the enclosed Amendment No. 38 to Facility Operating License No. NPF-37 and Amendment No. 38 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 25 to Facility Operating License No. NPF-72 and Amendment No. 25 to Facility Operating License No. NPF-77 for the Braidwood Station, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated January 31, 1990, as supplemented August 30, 1990.

These amendments approve changes to the Technical Specifications which would: (1) reduce the residual heat removal (RHR) minimum flowrate during refueling operations, (2) remove the RHR autoclosure interlock on the RHR system suction isolation valves, and (3) allow one safety injection pump to be available for injection purposes if normal heat removal capability were lost.

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,

/s/

Stephan P. Sands, 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. 38 to NPF-37
2. Amendment No. 38 to NPF-66
3. Amendment No. 25 to NPF-72
4. Amendment No. 25 to NPF-77
5. Safety Evaluation

cc w/enclosures:
See next page

DOCUMENT NAME: 76715/16/17/18 AMD

Office:	LA/PDIII-2	PM/PDIII-2	PM/PDIII-2	PD/PDIII-2	OGC-WF
Surname:	CMoore	SSands/tg	TBoyce	RBarrett	
Date:	8/24/90	8/24/90	8/24/90	8/31/90	8/30/90

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

August 31, 1990

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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,

Stephen P. Sands

Stephen P. Sands, 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. 38 to NPF-37
2. Amendment No. 38 to NPF-66
3. Amendment No. 25 to NPF-72
4. Amendment No. 25 to NPF-77
5. Safety Evaluation

cc w/enclosures:
See next page

Mr. Thomas J. Kovach
Commonwealth Edison Company

Byron/Braidwood Power Station
Unit Nos. 1 and 2

cc:
Mr. Gabe Toth
Westinghouse Electric Corporation
Energy Systems Business Unit
Post Office Box 355, Bay 236 West
Pittsburgh, Pennsylvania 15230

Joseph Gallo, Esq.
Hopkins and Sutter
888 16th Street, N.W.
Suite 700
Washington, D.C. 20006

Regional Administrator
U. S. NRC, Region III
799 Roosevelt Road, Bldg. #4
Glen Ellyn, Illinois 60137

Ms. Bridget Little Rorem
Appleseed Coordinator
117 North Linden Street
Essex, Illinois 60935

Mr. Edward R. Crass
Nuclear Safeguards and Licensing
Division
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
RR#1, Box 79
Braceville, Illinois 60407

Mr. Thomas W. Ortciger, Director
Illinois Emergency Services
and Disaster Agency
110 East Adams Street
Springfield, Illinois 62706

Robert Neumann
Office of Public Counsel
State of Illinois Center
100 W. Randolph
Suite 11-300
Chicago, Illinois 60601

U. S. Nuclear Regulatory Commission
Byron/Resident Inspectors Office
4448 North German Church Road
Byron, Illinois 61010

Ms. Lorraine Creek
Rt. 1, Box 182
Manteno, Illinois 60950

Mrs. Phillip B. Johnson
1907 Stratford Lane
Rockford, Illinois 61107

Douglass Cassel, Esq.
109 N. Dearborn Street
Suite 1300
Chicago, Illinois 60602

David C. Thomas, Esq.
77 S. Wacker Drive
Chicago, Illinois 60601

Michael Miller, Esq.
Sidley and Austin
One First National Plaza
Chicago, Illinois 60690

George L. Edgar
Newman & Holtzinger, P.C.
1615 L Street, N.W.
Washington, D.C. 20036

Commonwealth Edison Company
Byron Station Manager
4450 North German Church Road
Byron, Illinois 61010

Illinois Department of
Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. NPF-37

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 31, 1990, as supplemented August 30, 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. NPF-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 38 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, 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 31, 1990

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 38
License No. NPF-66

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 31, 1990, as supplemented August 30, 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 1;

   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. NPF-66 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 38 and revised by Attachment 2 to NPF-66, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

RJBarrett

Richard J. Barrett, 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 31, 1990

ATTACHMENT TO LICENSE AMENDMENT NOS. 38 AND 38

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. STN 50-454 AND STN 50-455

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 Pages	Insert Pages
IX	IX
X	X
XVI	XVI
XVIII	XVIII
3/4 4-41	3/4 4-41
3/4 5-5	3/4 5-5
3/4 5-9	3/4 5-9
-	3/4 5-10
-	3/4 5-11
3/4 9-9	3/4 9-9
3/4 9-10	3/4 9-10
B 3/4 4-16	B 3/4 4-16
B 3/4 5-1	B 3/4 5-1
B 3/4 5-2	B 3/4 5-2
-	B 3/4 5-3
-	B 3/4 5-4
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3

SECTION PAGE

SECTION PAGE

3/4.7 PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve RH8708B verify at least once per 72 hours that valves RH8702A and RH8702B are open.

b. For RHR suction relief valve RH8708A verify at least once per 72 hours that valves RH8701A and RH8701B are open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

SURVEILLANCE REQUIREMENTS (Continued)

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic interlock action of the RHR System from the Reactor Coolant System by ensuring that any simulated or actual Reactor Coolant System pressure signal greater than or equal to 360 psig prevents the valves from being opened.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal and on a RWST Level-Low-Low test signal, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1)	Centrifugal charging pump	$\geq$ 2396 psid,
2)	Safety Injection pump	$\geq$ 1412 psid, and
3)	RHR pump	In accordance with Figure 4.5-1

3/4.5.4 ECCS SUBSYSTEMS - T_{avg} LESS THAN OR EQUAL TO 200°F

PRESSURIZER LEVEL GREATER THAN 5 PERCENT (LEVEL 409.5')

LIMITING CONDITION FOR OPERATION

3.5.4.1 All Safety Injection pumps shall be inoperable.

APPLICABILITY: MODE 5 with pressurizer level greater than 5 percent, and MODE 6 with pressurizer level greater than 5 percent and the reactor vessel head resting on the reactor vessel flange.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS:

4.5.4.1 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position at least once per 12 hours.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge of the pump is isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

3/4.5.4 ECCS SUBSYSTEMS - T_{avg} LESS THAN OR EQUAL TO 200°F

PRESSURIZER LEVEL LESS THAN OR EQUAL TO 5 PERCENT (LEVEL 409.5')

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least one Safety Injection pump and flowpath shall be available,

or

the hot side of the RCS must be adequately vented and have valve alignments to allow gravity feed from the RWST.

APPLICABILITY: Either MODE 5 or MODE 6 with pressurizer level less than or equal to 5 percent.

ACTION:

If neither Safety Injection pump is available and the hot side of the RCS is not adequately vented then immediately initiate corrective action to restore either condition or establish pressurizer level greater than 5 percent.

SURVEILLANCE REQUIREMENTS

4.5.4.2.1 At least one Safety Injection pump shall be demonstrated available, when required, by verifying at least once per 12 hours that 1) the motor circuit breakers are racked in and open with the control switch in the pull out position, and 2) an OPERABLE flowpath exists from the RWST to the RCS, or

4.5.4.2.2 The RCS shall be demonstrated to be adequately vented, when required, by verifying at least once per 12 hours that:

a. One of the following hot side vent paths is available:

   1) The reactor vessel head is removed, or

   2) The pressurizer upper manway is removed, it has been at least 140 hours since shutdown and the RCS is 140°F or less, or

   3) Three pressurizer safety valves are removed, it has been at least 410 hours since shutdown and the RCS is 140°F or less, or

   4) Two pressurizer safety valves are removed, it has been at least 850 hours since shutdown and the RCS is 140°F or less.

b. An OPERABLE flowpath that will permit gravity feed from the RWST is available.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) and the heat traced portion of the RWST vent path shall be OPERABLE with:

a. A minimum contained borated water level of 89%,

b. A minimum boron concentration of 2000 ppm,

c. A minimum water temperature of 35°F, and

d. A maximum water temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water level in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 35°F or greater than 100°F, and

c. At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least once per 12 hours, one RHR loop shall be verified in operation and circulating coolant at a flowrate of greater than or equal to 1000 gpm with RCS temperature less than or equal to 140°F.

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least once per 12 hours one RHR loop shall be verified in operation and circulating coolant at a flowrate of greater than or equal to 1000 gpm with RCS temperature less than or equal to 140°F.

PRESSURE/TEMPERATURE LIMITS (Continued)

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or two RHR suction valves, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water solid RCS.

These two scenarios are analyzed to determine the resulting overshoots assuming a single PORV actuation with a stroke time of 2.0 seconds from full closed to full open. Figure 3.4-4 is based upon this analysis and represents the maximum allowable PORV variable setpoint such that, for the two overpressurization transients noted, the resulting pressure will not exceed the Appendix G reactor vessel NDT limits (nominal 10 effective full power years for Unit 1 only).

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. A contained borated water level between 31% and 63% ensures a volume of greater than or equal to 6995 gallons but less than or equal to 7217 gallons.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

The requirement to verify accumulator isolation valves shut with power removed from the valve operator when the pressurizer is solid ensures the accumulators will not inject water and cause a pressure transient when the Reactor Coolant System is on solid plant pressure control.

3/4.5.2, 3/4.5.3 AND 3/4.5.4 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE Charging pump to be inoperable in MODE 4 with one or more of the RCS cold legs less than or equal to 330°F, MODE 5, and MODE 6 with the reactor vessel head on, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or RHR suction relief valve. Similarly, the requirement to verify all Safety Injection pumps are inoperable in MODE 4 with the temperature of one or more of the RCS Cold Legs less than or equal to 330°F, in MODE 5 with pressurizer level greater than 5 percent (Level 409.5') and in MODE 6 with pressurizer level greater than 5 percent and the reactor vessel head resting on the reactor vessel flange, provides assurance that a mass addition pressure transient can be relieved by a single PORV or RHR suction relief valve.

In MODE 5 and MODE 6 with pressurizer level less than or equal to 5 percent, at least one Safety Injection pump or gravity feed from the RWST must be available to mitigate the effects of a loss of decay heat removal during partially drained conditions. Surveillance requirements assure availability, but prevent inadvertent actuation during these modes. The desired flow path for the SI pump or gravity feed varies with RCS configuration and is, therefore, procedurally addressed.

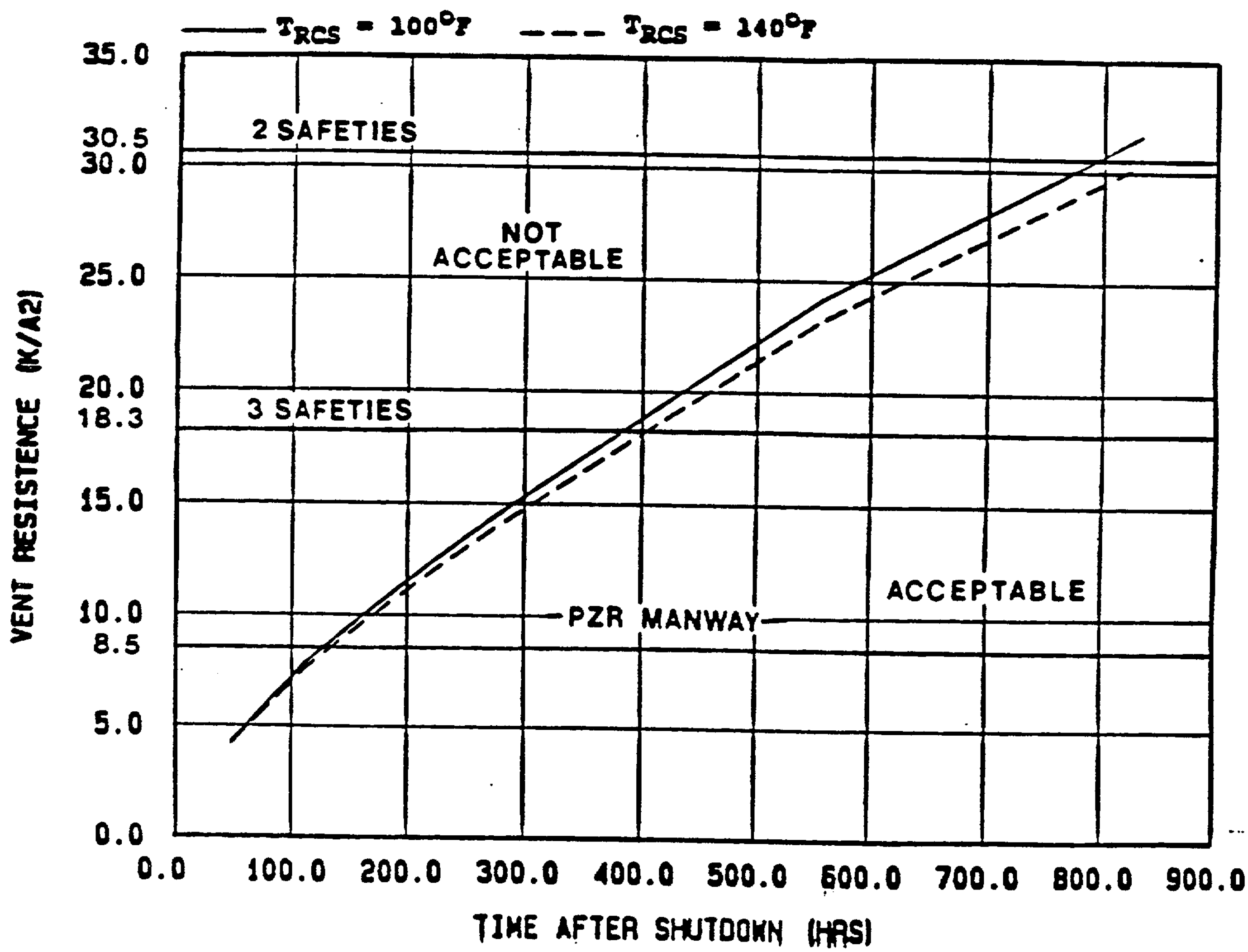
The Surveillance Requirements define what constitutes an adequate hot side vent for various plant conditions. It was determined that removing the reactor vessel head was an adequate vent under all conditions. Other venting alternatives have restrictions based on time from shutdown and RCS temperature. The values in the surveillance were taken from the graph on the following page.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The Surveillance Requirements for leakage testing of ECCS check valves ensures that a failure of one valve will not cause an intersystem LOCA. In Mode 3, with pressurizer pressure below 1000 psig, the accumulators will be available with their isolation valves either closed but energized, or open, whenever a SI8809 valve is closed to perform check valve leakage testing.

EMERGENCY CORE COOLING SYSTEMS

BASES

ECCS SUBSYSTEMS (Continued)



Vent Path Required to Prevent
RCS Pressurization

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. A minimum contained borated water level of 89% ensures a volume of greater than or equal to 395,000 gallons.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine and auxiliary hoist ensure that: (1) refueling machines will be used for movement of drive rods and fuel assemblies, (2) each refueling machine has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool areas ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The surveillance requirement verifies that the RHR loop is operating and circulating reactor coolant to ensure the capability of the RHR system to maintain compliance with plant design limits. The required RHR loop reactor coolant flowrate is determined by the flowrate necessary to: (1) provide sufficient decay heat removal capability, (2) maintain the reactor coolant temperature rise through the core within design limits, for compliance with flowrates assumed in the boron dilution analysis, (3) prevent thermal and boron stratification in the core, (4) preclude cavitation of the reactor coolant downstream of the RHR flow control valve, and (5) ensure that inadvertent boron dilution events can be identified and terminated by operator action prior to the reactor returning critical.

In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR loop flowrate determination must also consider the RHR pump suction requirements. At this water level, the RHR pump can experience vortexing or cavitation conditions which would cause the loss of RHR pump operation, if the flowrate demand is too high. Operation with reactor coolant water at this level is often called mid-loop operation. Care must be taken in determining the RHR loop flowrate, when operating with water level in this region, to prevent loss of the RHR pump and subsequent loss of the RHR loop for decay heat removal.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION (Continued)

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of RHR capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL HANDLING BUILDING EXHAUST FILTER PLENUM

The limitations on the Fuel Handling Building Exhaust Filter Plenum ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. NPF-72

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 31, 1990, as supplemented August 30, 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. NPF-72 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 25 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and is to be implemented by December 15, 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, 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 31, 1990

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 25
License No. NPF-77

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated January 31, 1990, as supplemented August 30, 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 1;

   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. NPF-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 25 and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-72, dated July 2, 1987, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and is to be implemented by December 15, 1990.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard J. Barrett, 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 31, 1990

ATTACHMENT TO LICENSE AMENDMENT NOS. 25 AND 25

FACILITY OPERATING LICENSE NOS. NPF-72 AND NPF-77

DOCKET NOS. STN 50-456 AND STN 50-457

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

Remove Pages	Insert Pages
IX	IX
X	X
XVI	XVI
XVIII	XVIII
3/4 4-41	3/4 4-41
3/4 5-5	3/4 5-5
3/4 5-9	3/4 5-9
-	3/4 5-10
-	3/4 5-11
3/4 9-9	3/4 9-9
3/4 9-10	3/4 9-10
B 3/4 4-16	B 3/4 4-16
B 3/4 5-1	B 3/4 5-1
B 3/4 5-2	B 3/4 5-2
-	B 3/4 5-3
-	B 3/4 5-4
B 3/4 9-2	B 3/4 9-2
B 3/4 9-3	B 3/4 9-3

SECTION PAGE

3/4.7 PLANT SYSTEMS

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.6 CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 Each RHR suction relief valve shall be demonstrated OPERABLE when the RHR suction relief valves are being used for cold overpressure protection as follows:

a. For RHR suction relief valve RH8708B verify at least once per 72 hours that valves RH8702A and RH8702B are open.

b. For RHR suction relief valve RH8708A verify at least once per 72 hours that valves RH8701A and RH8701B are open.

c. Testing pursuant to Specification 4.0.5.

4.4.9.3.3 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

SURVEILLANCE REQUIREMENTS (Continued)

1) For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY, and

2) Of the areas affected within containment at the completion of each containment entry when CONTAINMENT INTEGRITY is established.

d. At least once per 18 months by:

1) Verifying automatic interlock action of the RHR System from the Reactor Coolant System by ensuring that any simulated or actual Reactor Coolant System pressure signal greater than or equal to 360 psig prevents the valves from being opened.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. At least once per 18 months, during shutdown, by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal and on a RWST Level-Low-Low test signal, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying that each of the following pumps develops the indicated differential pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1)	Centrifugal charging pump	$\geq$ 2396 psid,
2)	Safety Injection pump	$\geq$ 1412 psid, and
3)	RHR pump	In accordance with Figure 4.5-1

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 ECCS SUBSYSTEMS - T_{avg} LESS THAN OR EQUAL TO 200°F

PRESSURIZER LEVEL GREATER THAN 5 PERCENT (LEVEL 409.5')

LIMITING CONDITION FOR OPERATION

3.5.4.1 All Safety Injection pumps shall be inoperable.

APPLICABILITY: MODE 5 with pressurizer level greater than 5 percent, and MODE 6 with pressurizer level greater than 5 percent and the reactor vessel head resting on the reactor vessel flange.

ACTION:

With a Safety Injection pump OPERABLE, restore all Safety Injection pumps to inoperable status within 4 hours.

SURVEILLANCE REQUIREMENTS:

4.5.4.1 All Safety Injection pumps shall be demonstrated inoperable* by verifying that the motor circuit breakers are secured in the open position at least once per 12 hours.

*An inoperable pump may be energized for testing or for filling accumulators provided the discharge of the pump is isolated from the RCS by a closed isolation valve with power removed from the valve operator, or by a manual isolation valve secured in the closed position.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 ECCS SUBSYSTEMS - T_{avg} LESS THAN OR EQUAL TO 200°F

PRESSURIZER LEVEL LESS THAN OR EQUAL TO 5 PERCENT (LEVEL 409.5')

LIMITING CONDITION FOR OPERATION

3.5.4.2 At least one Safety Injection pump and flowpath shall be available,

or

the hot side of the RCS must be adequately vented and have valve alignments to allow gravity feed from the RWST.

APPLICABILITY: Either MODE 5 or MODE 6 with pressurizer level less than or equal to 5 percent.

ACTION:

If neither Safety Injection pump is available and the hot side of the RCS is not adequately vented then immediately initiate corrective action to restore either condition or establish pressurizer level greater than 5 percent.

SURVEILLANCE REQUIREMENTS:

4.5.4.2.1 At least one Safety Injection pump shall be demonstrated available, when required, by verifying at least once per 12 hours that 1) the motor circuit breakers are racked in and open with the control switch in the pull out position, and 2) an OPERABLE flowpath exists from the RWST to the RCS, or

4.5.4.2.2 The RCS shall be demonstrated to be adequately vented, when required, by verifying at least once per 12 hours that:

a. One of the following hot side vent paths is available:

    1) The reactor vessel head is removed, or

    2) The pressurizer upper manway is removed, it has been at least 140 hours since shutdown and the RCS is 140°F or less, or

    3) Three pressurizer safety valves are removed, it has been at least 410 hours since shutdown and the RCS is 140°F or less, or

    4) Two pressurizer safety valves are removed, it has been at least 850 hours since shutdown and the RCS is 140°F or less.

b. An OPERABLE flowpath that will permit gravity feed from the RWST is available.

3/4.5.5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The refueling water storage tank (RWST) and the heat traced portion of the RWST vent path shall be OPERABLE with:

a. A minimum contained borated water level of 89%,

b. A minimum boron concentration of 2000 ppm,

c. A minimum water temperature of 35°F, and

d. A maximum water temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water level in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 35°F or greater than 100°F, and

c. At least once per 24 hours by verifying the RWST vent path temperature to be greater than or equal to 35°F when the outside air temperature is less than 35°F.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least once per 12 hours, one RHR loop shall be verified in operation and circulating coolant at a flowrate of greater than or equal to 1000 gpm with RCS temperature less than or equal to 140°F.

*The RHR loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least once per 12 hours one RHR loop shall be verified in operation and circulating coolant at a flowrate of greater than or equal to 1000 gpm with RCS temperature less than or equal to 140°F.

PRESSURE/TEMPERATURE LIMITS (Continued)

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, or two RHR suction valves, or an RCS vent opening of at least 2 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water solid RCS.

These two scenarios are analyzed to determine the resulting overshoots assuming a single PORV actuation with a stroke time of 2.0 seconds from full closed to full open. Figure 3.4-4 is based upon this analysis and represents the maximum allowable PORV variable setpoint such that, for the two overpressurization transients noted, the resulting pressure will not exceed the nominal 10 effective full power years (EFPY) Appendix G reactor vessel NDT limits.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. A contained borated water level between 31% and 63% ensures a volume of greater than or equal to 6995 gallons but less than or equal to 7217 gallons.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

The requirement to verify accumulator isolation valves shut with power removed from the valve operator when the pressurizer is solid ensures the accumulators will not inject water and cause a pressure transient when the Reactor Coolant System is on solid plant pressure control.

3/4.5.2, 3/4.5.3 AND 3/4.5.4 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

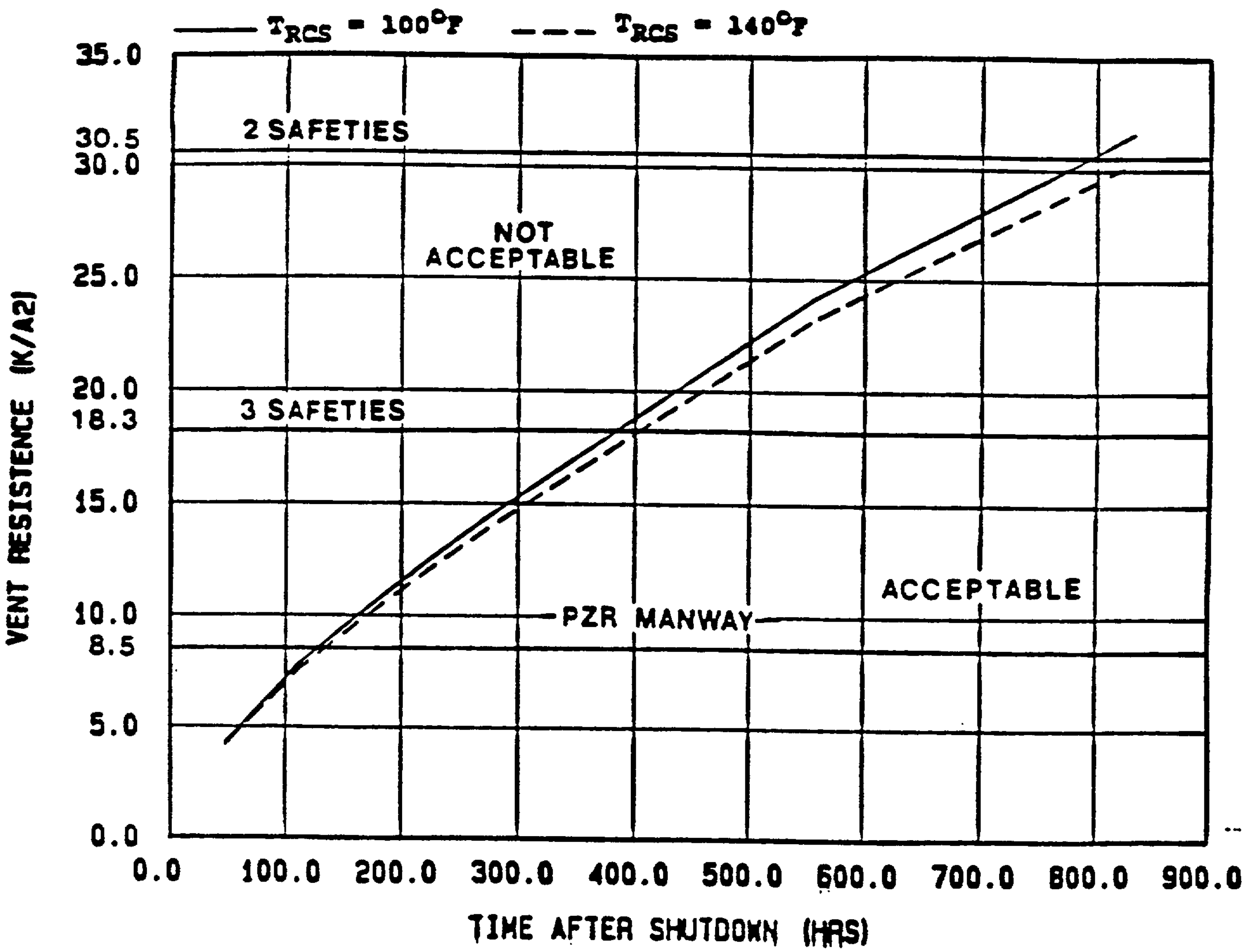
The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE Charging pump to be inoperable in MODE 4 with one or more of the RCS cold legs less than or equal to 330°F, MODE 5, and MODE 6 with the reactor vessel head on, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or RHR suction relief valve. Similarly, the requirement to verify all Safety Injection pumps are inoperable in MODE 4 with the temperature of one or more of the RCS Cold Legs less than or equal to 330°F, in MODE 5 with pressurizer level greater than 5 percent (Level 409.5') and in MODE 6 with pressurizer level greater than 5 percent and the reactor vessel head resting on the reactor vessel flange, provides assurance that a mass addition pressure transient can be relieved by a single PORV or RHR suction relief valve.

In MODE 5 and MODE 6 with pressurizer level less than or equal to 5 percent, at least one Safety Injection pump or gravity feed from the RWST must be available to mitigate the effects of a loss of decay heat removal during partially drained conditions. Surveillance requirements assure availability, but prevent inadvertent actuation during these modes. The desired flow path for the SI pump or gravity feed varies with RCS configuration and is, therefore, procedurally addressed.

The Surveillance Requirements define what constitutes an adequate hot side vent for various plant conditions. It was determined that removing the reactor vessel head was an adequate vent under all conditions. Other venting alternatives have restrictions based on time from shutdown and RCS temperature. The values in the surveillance were taken from the graph on the following page.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The Surveillance Requirements for leakage testing of ECCS check valves ensures that a failure of one valve will not cause an intersystem LOCA. In Mode 3, with pressurizer pressure below 1000 psig, the accumulators will be available with their isolation valves either closed but energized, or open, whenever a SI8809 valve is closed to perform check valve leakage testing.

ECCS SUBSYSTEMS (Continued)



Vent Path Required to Prevent
RCS Pressurization

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. A minimum contained borated water level of 89% ensures a volume of greater than or equal to 395,000 gallons.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine and auxiliary hoist ensure that: (1) refueling machines will be used for movement of drive rods and fuel assemblies, (2) each refueling machine has sufficient load capacity to lift a drive rod or fuel assembly, and (3) the core internals and reactor vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE FACILITY

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool areas ensures that in the event this load is dropped: (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The surveillance requirement verifies that the RHR loop is operating and circulating reactor coolant to ensure the capability of the RHR system to maintain compliance with plant design limits. The required RHR loop reactor coolant flowrate is determined by the flowrate necessary to: (1) provide sufficient decay heat removal capability, (2) maintain the reactor coolant temperature rise through the core within design limits, for compliance with flowrates assumed in the boron dilution analysis, (3) prevent thermal and boron stratification in the core, (4) preclude cavitation of the reactor coolant downstream of the RHR flow control valve, and (5) ensure that inadvertent boron dilution events can be identified and terminated by operator action prior to the reactor returning critical.

In addition, during operation of the RHR loop with the water level in the vicinity of the reactor vessel nozzles, the RHR loop flowrate determination must also consider the RHR pump suction requirements. At this water level, the RHR pump can experience vortexing or cavitation conditions which would cause the loss of RHR pump operation, if the flowrate demand is too high. Operation with reactor coolant water at this level is often called mid-loop operation. Care must be taken in determining the RHR loop flowrate, when operating with water level in this region, to prevent loss of the RHR pump and subsequent loss of the RHR loop for decay heat removal.

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION (Continued)

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of RHR capability. With the reactor vessel head removed and at least 23 feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT PURGE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge penetrations will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 FUEL HANDLING BUILDING EXHAUST FILTER PLENUM

The limitations on the Fuel Handling Building Exhaust Filter Plenum ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 38 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 25 TO FACILITY OPERATING LICENSE NO. NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNIT NOS. 1 AND 2

BRAIDWOOD STATION, UNIT NOS. 1 AND 2

DOCKET NOS. STN 50-454, STN 50-455, STN 50-456 AND STN 50-457

1.0 INTRODUCTION

By letter dated January 31, 1990, as supplemented August 30, 1990, Commonwealth Edison Company (CECo), the licensee, submitted proposed amendment changes to the Technical Specifications (TS) and associated bases for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. The proposal is a result of an engineering review performed to identify any changes that might mitigate or prevent the consequences of a loss of decay heat removal event. The licensee has identified, in response to Generic Letter 88-17, three changes: (1) reduction of the Residual Heat Removal (RHR) minimum flowrate during refueling operations; (2) removal of the RHR auto closure interlock on the RHR system suction isolation valves; (3) allowance for one safety injection (SI) pump to be available for injection during low temperature operation.

Based on our review of the licensee's submittal, we have found the revisions to be acceptable.

2.0 MINIMUM RHR FLOWRATE

The required minimum RHR flowrate during midloop operations is based on the ability of the RHR to remove decay heat such that Reactor Coolant System (RCS) temperature can be controlled and that the reactor coolant temperature rise through the core does not exceed reactor vessel internals delta T limits. The flow rate ensures that there is sufficient coolant circulation maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification, and that the pressure drop across the RHR bypass flow control valve does not result in cavitation.

Vortexing at the junction of the RHR system suction line and the RCS may occur if the water level is too low, a situation to be avoided since this may introduce air into the RHR system pump suction. Vortexing can occur more easily when flow is high. The likelihood of vortex formation due to partial draining of the RCS can be offset by reducing the RHR flow rate. As discussed in NUREG-1269, "Loss of Residual Heat Removal System, Diablo Canyon, Unit 2, April 10, 1987," reduced RHR system flow rate would provide a greater margin against vortexing and can help preclude an inadvertent loss of decay heat removal (DHR) capability due to air entrainment and cavitation of the RHR system pumps.

The licensee has revised Specification 4.9.8.1 to reflect a reduction in flow rate from "greater than or equal to 2800 gpm" to "greater than or equal to 1000 gpm." As long as one RHR pump is running, even at flow rates as low as 1000 gpm, there is enough mixing in the flow to ensure that no boron stratification occurs, as well as the uniform distribution of boron to minimize the effects of a boron dilution scenario. This reduction in flow rate will reduce the susceptibility to vortexing, and is consistent with NUREG-1269. However, 1000 gpm is not always sufficient to maintain the RCS temperature less than or equal to 140 degrees F, for example, shortly after shutdown. Hence, the change proposed by the licensee will include the addition of a Surveillance Requirement which "verifies that the RCS temperature is being maintained at less than or equal to 140 degrees F" while operating at the reduced flowrate. This is consistent with Mode 6 operating requirements.

Since the proposed Technical Specification will require an RHR flowrate that provides adequate mixing and is sufficient to maintain RCS temperature less than 140 degrees F, the staff finds the proposed Technical Specification modifications to be acceptable.

2.1 Removal of the Auto Closure Interlock

During normal and emergency conditions, the low pressure RHR system is isolated from the high pressure RCS. This isolation is necessary to avoid damages resulting from overpressurization, and minimize the potential for loss of integrity of the low pressure system and possible radioactive releases to the environment. The purpose of the Auto Closure Interlock (ACI) is to preclude conditions that could lead to an interfacing system loss-of-coolant accident (LOCA) by ensuring that both suction/isolation valves in each RHR system train are fully closed when the reactor coolant system is pressurized above the RHR design pressure.

There is a competing concern regarding the ACI however, and that is the potential for the ACI circuitry to cause inadvertent RHR isolation during cold and refueling operations. Westinghouse performed a generic evaluation (WCAP - 11736) to study the impact of removing the ACI feature. The results of the evaluation showed that removal of the RHR ACI combined with the addition of alarms and procedures improves the availability of the RHR system during short-term and long-term cooldown and reduces the estimated core damage frequency. The licensee has, therefore, proposed to remove the ACI at the Byron and Braidwood Stations and substitute alternative features to protect against RHR overpressurization. These changes are consistent with the staff's position taken on the removal of the ACI. The changes consist of hardware and procedural enhancements that the staff believes will produce a net safety benefit. The hardware changes include the addition of an alarm to each RHR suction valve. The setpoint for the alarm will be within the range of the open permissive setpoint pressure and the RHR system design pressure minus the RHR pump head. The open permissive that prevents these valves from being opened will be left in place and will not be disabled by the addition of the alarm and the removal of the ACI circuitry. The valve position indicator to the alarm will not be affected by power lockout of the RHR suction valves and a method independent of the alarm for determining valve position will be available in the control room following power lockout of the RHR suction valves. The licensee has committed to the hardware changes listed above and has proposed two methods independent of the alarm for determining valve position:

(a) Normal position indication powered from an independent valve supply source;

(b) Monitor Light Group 1, powered from an annunciator input cabinet (an independent power source) and located on the control room panel;

The licensee has also committed to the following procedural modifications:

(1) The alarm response procedure used during plant startup will be modified to reflect alarm recognition responses for the added alarm. The procedure will be revised to direct the operator to take the necessary actions to close the open RHR suction valve(s), if they are not closed following alarm actuation. If this is not possible, the operator will be instructed to not pressurize further and to return to the safe shutdown mode of operation.

(2) A surveillance procedure for the RHR suction valve alarms is added to periodically verify that the alarms are operable.

(3) A method independent of the alarm will be used to verify that these valves are closed when the power to these valves is locked out.

In a teleconference held July 24, 1990, between the licensee and the NRC, the licensee also stated that an instrument surveillance procedure was in place to periodically verify operability of the alarms.

The staff has reviewed the Byron and Braidwood Stations submittal and has found the proposed hardware changes and procedural modifications are adequate alternatives to assure RHR system overpressure protection. Removal of the ACI is, therefore, acceptable.

2.2 Safety Injection Pump Availability

The availability of a Safety Injection (SI) pump provides for the mitigation of the effects of a loss of decay heat removal event during mid-loop operations. Operation of at least one SI pump is required in some cases to prevent core uncovery. The licensee proposes to have an SI pump available in Modes 5 and 6. The potential for low temperature overpressurization has been analyzed and accounted for in the Specification by requiring pressurizer level to be less than 5 percent if the SI pump is available. It is the licensee's intention that during RCS reduced inventory conditions, the safety injection pump motor circuit breakers will be racked in and the pump secured by placing the Control Room handswitch in the Pull-to-Lock position. This will prevent the safety injection pump from being inadvertently started by a signal, but will allow the operators to start the pump from the Control Room if needed to mitigate a loss of decay heat removal.

This modification is consistent with Generic Letter 88-17 and is acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of the facility components located within the restricted areas as defined in 10 CFR Part 20 or a change to a surveillance requirement. The staff has determined that these 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 previously issued a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding. 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 or environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: A. Massey

Dated: August 31, 1990