50-454/455
50-456/457

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

January 23, 1998

Mr. Oliver D. Kingsley, President
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M98799, M98800, M98801 AND M98802)

Dear Mr. Kingsley:

The U.S. Nuclear Regulatory Commission (Commission) has issued the enclosed Amendment No. 98 to Facility Operating License No. NPF-37 and Amendment No. 98 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively, and Amendment No. 89 to Facility Operating License No. NPF-72 and Amendment No. 89 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 May 21, 1997, as supplemented by letters dated November 18, 1997, December 3, 1997, January 8, 1998 and January 13, 1998.

The amendments relocate the reactor coolant system pressure and temperature limits for heatup, cooldown, low-temperature operation and hydrostatic testing, and the associated low-temperature overpressure protection (LTOP) system setpoint curves into a Pressure and Temperature Limits Report (PTLR).

In addition, by letter of May 6, 1997, as supplemented by the December 3, 1997, letter, you requested permission to integrate the reactor vessel weld metal surveillance data between Byron, Units 1 and 2, and Braidwood, Units 1 and 2. The Commission approved this request, in principle, by letter dated January 16, 1998; however, the limitations on data integration are addressed in the approved methodology for developing the PTLR.

DF01

NRC FILE CENTER COPY

9802050177 980123
PDR ADOCK 05000454
P PDR

CP-1

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,

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 98 to NPF-37
2. Amendment No. 98 to NPF-66
3. Amendment No. 89 to NPF-72
4. Amendment No. 89 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

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,

Orig. signed by

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

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

Enclosures:
1. Amendment No. 98 to NPF-37
2. Amendment No. 98 to NPF-66
3. Amendment No. 89 to NPF-72
4. Amendment No. 89 to NPF-77
5. Safety Evaluation

cc w/encls: See next page

DISTRIBUTION:

Docket File	PUBLIC
PDIII-2 r/f (2)	C. Moore (2)
E. Adensam, EGA1	T. Harris, TLH3
R. Capra	R. Assa
G. Dick (3)	OGC, O15B18
S. Bailey	W. Beckner, O13H15
G. Hill (8), T5C3	M. Jordan, RIII
ACRS, T2E26	

DOCUMENT NAME: G:\CMNTSP\BRAID-BY\BB98800.AMD

To receive a copy of this document, indicate in the box: "C" = Copy without enclosures "E" = Copy with enclosures "N" = No copy

OFFICE	PM:PDIII-2 E	LA:PDIII-2	OGC	D:PDIII-2	PM:PDIII-2
NAME	GDICK	CMOORE	R Bachmann	RCAPRA	SBAILEY
DATE	01/23/98	01/23/98	01/21/98	01/23/98	01/23/98

OFFICIAL RECORD COPY

cc:

Mr. William P. Poirier, Director
Westinghouse Electric Corporation
Energy Systems Business Unit
Post Office Box 355, Bay 236 W.
Pittsburgh, Pennsylvania 15230

Joseph Gallo
Gallo & Ross
1250 Eye St., N.W., Suite 302
Washington, DC 20005

Michael I. Miller, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Howard A. Learner
Environmental law and Policy
Center of the Midwest
203 N. LaSalle St. Suite 1390
Chicago, Illinois 60601

U.S. Nuclear Regulatory Commission
Byron Resident Inspectors Office
4448 N. German Church Road
Byron, Illinois 61010-9750

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

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

Chairman, Ogle County Board
Post Office Box 357
Oregon, Illinois 61061

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

George L. Edgar
Morgan, Lewis and Bochius
1800 M Street, N.W.
Washington, DC 20036

Attorney General
500 S. Second Street
Springfield, Illinois 62701

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

Commonwealth Edison Company
Byron Station Manager
4450 N. German Church Road
Byron, Illinois 61010-9794

Commonwealth Edison Company
Site Vice President - Byron
4450 N. German Church Road
Byron, Illinois 61010-9794

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

Mr. Ron Stephens
Illinois Emergency Services
and Disaster Agency
110 E. Adams Street
Springfield, Illinois 62706

Chairman
Will County Board of Supervisors
Will County Board Courthouse
Joliet, Illinois 60434

Commonwealth Edison Company
Braidwood Station Manager
RR 1, Box 84
Braceville, Illinois 60407

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

Document Control Desk-Licensing
Commonwealth Edison Company
1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

Commonwealth Edison Company
Site Vice President - Braidwood
RR 1, Box 84
Braceville, IL 60407

Mr. Michael J. Wallace
Nuclear Services Senior Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, IL 60515

Mr. Gene H. Stanley
PWR's Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, IL 60515

Mr. Steve Perry
BWR's Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, IL 60515

Mr. Dennis Farrar
Regulatory Services Manager
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

Ms. Irene Johnson, Licensing Director
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

Commonwealth Edison Company
Reg. Assurance Supervisor - Braidwood
RR 1, Box 79
Braceville, Illinois 60407

Commonwealth Edison Company
Reg. Assurance Supervisor - Byron
4450 N. German Church Road
Byron, Illinois 61010-9794

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-454

BYRON STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
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 May 21, 1997, as supplemented by letters dated November 18, 1997, December 3, 1997, January 8, 1998 and January 13, 1998, 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:

9802050194 980123
PDR ADOCK 05000454
P PDR

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 98 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 shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick, Jr.

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 23, 1998

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-455

BYRON STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 98
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 May 21, 1997, as supplemented by letters dated November 18, 1997, December 3, 1997, January 8, 1998 and January 13, 1998, 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-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. 98 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 and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick Jr.

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 23, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 98 AND 98

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
I	I
VIII	VIII
XV	XV
--	1-4a
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
3/4 4-34	3/4 4-34
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
3/4 4-37	3/4 4-37
3/4 4-38	--
3/4 4-39	--
3/4 4-40a	--
3/4 4-40b	--
3/4 4-41	--
3/4 4-42	--
3/4 4-43	--
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	--
B 3/4 4-12	--
B 3/4 4-13	--
B 3/4 4-14	--
B 3/4 4-15	--
B 3/4 4-16	--
B 3/4 4-17	--
6-22b	6-22b

INDEX

DEFINITIONS

SECTION	PAGE
1.0 DEFINITIONS	
1.1 ACTION	1-1
1.2 ACTUATION LOGIC TEST	1-1
1.3 ANALOG CHANNEL OPERATIONAL TEST	1-1
1.4 AXIAL FLUX DIFFERENCE	1-1
1.5 CHANNEL CALIBRATION	1-1
1.6 CHANNEL CHECK	1-1
1.7 CONTAINMENT INTEGRITY	1-2
1.8 CONTROLLED LEAKAGE	1-2
1.9 CORE ALTERATION	1-2
1.10 DIGITAL CHANNEL OPERATIONAL TEST	1-2
1.11 DOSE EQUIVALENT I-131	1-2a
1.12 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY	1-3
1.13 ENGINEERED SAFETY FEATURES RESPONSE TIME	1-3
1.14 FREQUENCY NOTATION	1-3
1.15 IDENTIFIED LEAKAGE	1-3
1.15.a L_a	1-3
1.16 MASTER RELAY TEST	1-3
1.17 MEMBER(S) OF THE PUBLIC	1-3
1.18 OFFSITE DOSE CALCULATION MANUAL	1-4
1.19 OPERABLE - OPERABILITY	1-4
1.19.a OPERATING LIMITS REPORT	1-4
1.20 OPERATIONAL MODE - MODE	1-4
1.20.a P_a	1-4
1.21 PHYSICS TESTS	1-4
1.22 PRESSURE BOUNDARY LEAKAGE	1-4
1.22.a PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	1-4a
1.23 PROCESS CONTROL PROGRAM	1-5
1.24 PURGE - PURGING	1-5
1.25 QUADRANT POWER TILT RATIO	1-5
1.26 RATED THERMAL POWER	1-5
1.27 REACTOR TRIP SYSTEM RESPONSE TIME	1-5
1.28 REPORTABLE EVENT	1-5

SECTION PAGE

TABLE 3.4-2 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS..... 3/4 4-25

TABLE 4.4-3 REACTOR COOLANT SYSTEM CHEMISTRY SURVEILLANCE REQUIREMENTS..... 3/4 4-26

3/4.4.8 SPECIFIC ACTIVITY..... 3/4 4-27

FIGURE 3.4-1 DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1μCi/GRAM DOSE EQUIVALENT I-131..... 3/4 4-29

TABLE 4.4-4 REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM..... 3/4 4-30

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

Reactor Coolant System..... 3/4 4-32

Pressurizer..... 3/4 4-33

Overpressure Protection Systems..... 3/4 4-34

3/4.4.10 STRUCTURAL INTEGRITY..... 3/4 4-36

3/4.4.11 REACTOR COOLANT SYSTEM VENTS..... 3/4 4-37

SECTION PAGE

3/4.4.5 STEAM GENERATORS..... B 3/4 4-3

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE..... B 3/4 4-4

3/4.4.7 CHEMISTRY..... B 3/4 4-5

3/4.4.8 SPECIFIC ACTIVITY..... B 3/4 4-5

3/4.4.9 PRESSURE/TEMPERATURE LIMITS..... B 3/4 4-7

3/4.4.10 STRUCTURAL INTEGRITY..... B 3/4 4-10

3/4.4.11 REACTOR VESSEL HEAD VENTS..... B 3/4 4-10

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS..... B 3/4 5-1

3/4.5.2, 3/4.5.3 and 3/4.5.4 ECCS SUBSYSTEMS..... B 3/4 5-1

3/4.5.5 REFUELING WATER STORAGE TANK..... B 3/4 5-4

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT..... B 3/4 6-1

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS..... B 3/4 6-3

3/4.6.3 CONTAINMENT ISOLATION VALVES..... B 3/4 6-4

3/4.6.4 COMBUSTIBLE GAS CONTROL..... B 3/4 6-4

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.22.a The PTLR is the unit-specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the pressurizer power operated relief valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.1.11. Unit operation within these limits is addressed in LCO 3.4.9.1, "Pressure/Temperature Limits," and LCO 3.4.9.3, "Overpressure Protection Systems."

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System temperature and pressure, and heatup and cooldown rates shall be maintained in accordance with the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTION:

With any of the limits in the PTLR exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in the PTLR.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least two overpressure protection devices shall be OPERABLE, and each device shall be either:

a. A residual heat removal (RHR) suction relief valve with a lift setting of less than or equal to 450 psig, or

b. A power operated relief valve (PORV) with a lift setpoint that varies with RCS temperature which does not exceed the limit established in the PTLR.

APPLICABILITY: MODES 4, 5, and 6 with the reactor vessel head on.

ACTION:

a. With one of the two required overpressure protection devices inoperable in MODE 4, restore two overpressure protection devices to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With one of the two required overpressure protection devices inoperable in MODES 5 or 6, restore two overpressure protection devices to OPERABLE status within 24 hours or vent the RCS through at least a 2 square inch vent within the next 8 hours.

c. With both of the required overpressure protection devices inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

e. In the event either the PORVs, RHR suction relief valves, or the RCS vents are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, RHR suction relief valves, or RCS vents on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORVs are being used for cold overpressure protection by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days when the PORV is required OPERABLE; and

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.

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.

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected as follows:

a. Volumetric examination of the areas of higher stress concentration at the bore and keyways will be performed each 40 month period during refueling or maintenance shutdowns coinciding with the service inspection schedule as required by Section XI of the ASME Code.

b. Visual examination of all exposed surfaces will be performed and a surface examination of the bore and keyway surfaces will be performed whenever the flywheels are removed for maintenance purposes, but not more frequently than once each 10 year interval.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor vessel head vent path consisting of two valves in series powered from emergency busses shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the above reactor vessel head vent path inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING, and

c. Verifying flow through the reactor vessel head vent paths during venting operations at COLD SHUTDOWN or REFUELING.

SPECIFIC ACTIVITY (Continued)

take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomenon. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. LCO 3.4.9.1, "Pressure/Temperature Limits," limits the pressure and temperature changes during RCS heatup and cooldown to within the design assumptions and the stress limits for cyclic operation.

The PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) contains pressure and temperature (P/T) limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

LCO 3.4.9.1 establishes operating limits that provide a margin to non-ductile failure of the reactor vessel and piping of the Reactor Coolant Pressure Boundary (RCPB). The vessel is the component most subject to non-ductile failure, and the LCO limits apply to the entire RCS, except the pressurizer, which has different design characteristics and operating functions.

10 CFR Part 50, Appendix G, requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Appendix G of 10 CFR Part 50 requires an adequate margin to non-ductile failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G.

The neutron embrittlement effect on the material toughness is reflected by increasing the Nil Ductility Reference Temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 and Appendix H of 10 CFR Part 50. The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99, Revision 2.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate

PRESSURE/TEMPERATURE LIMITS (Continued)

the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls during heatup and cooldown, respectively.

The criticality limit curve includes the 10 CFR Part 50, Appendix G, requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.1.1.4, "Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in non-ductile failure of the RCPB, possibly leading to a nonisolable leak or loss-of-coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E, provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

OVERPRESSURE PROTECTION SYSTEMS

The Low Temperature Overpressure Protection (LTOP) System controls the RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the P/T limits of 10 CFR Part 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the pressurizer Power-Operated Relief Valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the 10 CFR Part 50, Appendix G, requirements during the MODES in which the LTOP system is necessary.

The reactor vessel material is less ductile at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and the material becomes less resistant to stress at low temperatures. RCS pressure, therefore, is maintained low at low temperatures and is increased only within the limits specified in the PTLR.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only during shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause non-ductile failure of the reactor vessel. LCO 3.4.9.1, "Pressure/Temperature Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

LCO 3.4.9.3, "Overpressure Protection Systems," provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all Safety Injection (SI) pumps and all but one charging pump (a centrifugal

PRESSURE/TEMPERATURE LIMITS (Continued)

charging pump) incapable of injection into the RCS and isolation of the SI accumulators. The pressure relief capacity requires either two redundant RCS relief valves, or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one centrifugal charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The LTOP System for pressure relief consists of two PORVs with reduced lift settings, or two Residual Heat Removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve, or a depressurized RCS and a RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to prevent overpressurization for the required coolant input capability.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The LTOP actuation logic monitors both RCS temperature and RCS pressure and determines when a condition is approaching the PTLR limits. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The PTLR presents the PORV setpoints for the LTOP system. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the 10 CFR Part 50, Appendix G, limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the subsequent relief will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RHR Suction Relief Valve Requirements

During the LTOP MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot legs to the inlets of the RHR pumps.

PRESSURE/TEMPERATURE LIMITS (Continued)

While these valves are open, the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.

The RHR suction isolation valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. The RHR suction relief valves are spring loaded, bellows-type relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for Class 2 relief valves.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve; removing a PORV's internals, and disabling its block valve in the open position; or similarly establishing any comparable vent. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

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).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.11 REACTOR VESSEL HEAD VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of a reactor vessel head vent path ensures the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System Vent Systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

REPORTING REQUIREMENTS (Continued)

OPERATING LIMITS REPORT (Continued)

9. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," dated August 1985 (Westinghouse Proprietary). (Methodology for Specification: Axial Flux Difference, Heat Flux Hot Channel Factor, and Nuclear Enthalpy Rise Hot Channel Factor).

10. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Controls Systems."

The operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

The OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

6.9.1.10 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates and pressurizer power-operated relief valve lift settings, shall be established and documented in the PTLR for the following:

1) LCO 3.4.9.1, "Pressure/Temperature Limits," and
2) LCO 3.4.9.3, "Overpressure Protection Systems."

The analytical methods used to determine the RCS pressure and temperature limits shall be those reviewed and approved by the NRC, specifically those described in NRC letter dated January 21, 1998, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report."

The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-456

BRAIDWOOD STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
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 May 21, 1997, as supplemented by letters dated November 18, 1997, December 3, 1997, January 8, 1998 and January 13, 1998, 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. 89 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 shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 23, 1998

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. STN 50-457

BRAIDWOOD STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 89
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 May 21, 1997, as supplemented by letters dated November 18, 1997, December 3, 1997, January 8, 1998 and January 13, 1998, 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-77 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 89 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 if its issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

George F. Dick, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 23, 1998

ATTACHMENT TO LICENSE AMENDMENT NOS. 89 AND 89

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
I	I
VIII	VIII
XV	XV
--	1-4a
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
3/4 4-34	3/4 4-34
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
3/4 4-37	3/4 4-37
3/4 4-38	--
3/4 4-39	--
3/4 4-40	--
3/4 4-40a	--
3/4/4-41	--
3/4 4-42	--
3/4 4-43	--
B 3/4 4-7	B 3/4 4-7
B 3/4 4-8	B 3/4 4-8
B 3/4 4-9	B 3/4 4-9
B 3/4 4-10	B 3/4 4-10
B 3/4 4-11	B 3/4 4-11
B 3/4 4-12	--
B 3/4 4-13	--
B 3/4 4-14	--
B 3/4 4-15	--
B 3/4 4-16	--
B 3/4 4-17	--
6-22b	6-22b

INDEX

DEFINITIONS

SECTION	PAGE
1.0 DEFINITIONS	
1.1 ACTION	1-1
1.2 ACTUATION LOGIC TEST	1-1
1.3 ANALOG CHANNEL OPERATIONAL TEST	1-1
1.4 AXIAL FLUX DIFFERENCE	1-1
1.5 CHANNEL CALIBRATION	1-1
1.6 CHANNEL CHECK	1-1
1.7 CONTAINMENT INTEGRITY	1-2
1.8 CONTROLLED LEAKAGE	1-2
1.9 CORE ALTERATION	1-2
1.10 DIGITAL CHANNEL OPERATIONAL TEST	1-2
1.11 DOSE EQUIVALENT I-131	1-2
1.12 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY	1-3
1.13 ENGINEERED SAFETY FEATURES RESPONSE TIME	1-3
1.14 FREQUENCY NOTATION	1-3
1.15 IDENTIFIED LEAKAGE	1-3
1.15.a L_a	1-3
1.16 MASTER RELAY TEST	1-3
1.17 MEMBER(S) OF THE PUBLIC	1-3
1.18 OFFSITE DOSE CALCULATION MANUAL	1-4
1.19 OPERABLE - OPERABILITY	1-4
1.19.a OPERATING LIMITS REPORT	1-4
1.20 OPERATIONAL MODE - MODE	1-4
1.20.a P_a	1-4
1.21 PHYSICS TESTS	1-4
1.22 PRESSURE BOUNDARY LEAKAGE	1-4
1.22.a PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	1-4a
1.23 PROCESS CONTROL PROGRAM	1-5
1.24 PURGE - PURGING	1-5
1.25 QUADRANT POWER TILT RATIO	1-5
1.26 RATED THERMAL POWER	1-5
1.27 REACTOR TRIP SYSTEM RESPONSE TIME	1-5
1.28 REPORTABLE EVENT	1-5

SECTION	PAGE
TABLE 3.4-2 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS	3/4 4-25
TABLE 4.4-3 REACTOR COOLANT SYSTEM CHEMISTRY SURVEILLANCE REQUIREMENTS	3/4 4-26
3/4.4.8 SPECIFIC ACTIVITY	3/4 4-27
FIGURE 3.4-1 DOSE EQUIVALENT I-131 REACTOR COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE REACTOR COOLANT SPECIFIC ACTIVITY >1μCI/GRAM DOSE EQUIVALENT I-131	3/4 4-29
TABLE 4.4-4 REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM	3/4 4-30
3/4.4.9 PRESSURE/TEMPERATURE LIMITS	
Reactor Coolant System	3/4 4-32
Pressurizer	3/4 4-33
Overpressure Protection Systems	3/4 4-34
3/4.4.10 STRUCTURAL INTEGRITY	3/4 4-36
3/4.4.11 REACTOR COOLANT SYSTEM VENTS	3/4 4-37

SECTION		PAGE
3/4.4.5	STEAM GENERATORS	B 3/4 4-3
3/4.4.6	REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-4
3/4.4.7	CHEMISTRY	B 3/4 4-5
3/4.4.8	SPECIFIC ACTIVITY	B 3/4 4-5
3/4.4.9	PRESSURE/TEMPERATURE LIMITS	B 3/4 4-7
3/4.4.10	STRUCTURAL INTEGRITY	B 3/4 4-10
3/4.4.11	REACTOR VESSEL HEAD VENTS	B 3/4 4-10

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1	ACCUMULATORS	B 3/4 5-1
3/4.5.2, 3/4.5.3 and 3/4.5.4	ECCS SUBSYSTEMS	B 3/4 5-1
3/4.5.5	REFUELING WATER STORAGE TANK	B 3/4 5-4

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1	PRIMARY CONTAINMENT	B 3/4 6-1
3/4.6.2	DEPRESSURIZATION AND COOLING SYSTEMS	B 3/4 6-3
3/4.6.3	CONTAINMENT ISOLATION VALVES	B 3/4 6-4
3/4.6.4	COMBUSTIBLE GAS CONTROL	B 3/4 6-4

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

1.22.a The PTLR is the unit-specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, and the pressurizer power operated relief valve (PORV) lift settings for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.1.11. Unit operation within these limits is addressed in LCO 3.4.9.1, "Pressure/Temperature Limits," and LCO 3.4.9.3, "Overpressure Protection Systems."

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System temperature and pressure, and heatup and cooldown rates shall be maintained in accordance with the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTION:

With any of the limits in the PTLR exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in the PTLR.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during auxiliary spray operation.

LIMITING CONDITION FOR OPERATION

3.4.9.3 At least two overpressure protection devices shall be OPERABLE, and each device shall be either:

a. A residual heat removal (RHR) suction relief valve with a lift setting of less than or equal to 450 psig, or

b. A power operated relief valve (PORV) with a lift setpoint that varies with RCS temperature which does not exceed the limit established in the PTLR.

APPLICABILITY: MODES 4, 5, and 6 with the reactor vessel head on.

ACTION:

a. With one of the two required overpressure protection devices inoperable in MODE 4, restore two overpressure protection devices to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2 square inch vent within the next 8 hours.

b. With one of the two required overpressure protection devices inoperable in MODES 5 or 6, restore two overpressure protection devices to OPERABLE status within 24 hours or vent the RCS through at least a 2 square inch vent within the next 8 hours.

c. With both of the required overpressure protection devices inoperable, depressurize and vent the RCS through at least a 2 square inch vent within 8 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

e. In the event either the PORVs, RHR suction relief valves, or the RCS vents are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, RHR suction relief valves, or RCS vents on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE when the PORVs are being used for cold overpressure protection by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days when the PORV is required OPERABLE; and

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.

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.

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10 In addition to the requirements of Specification 4.0.5, each reactor coolant pump flywheel shall be inspected as follows:

a. Volumetric examination of the areas of higher stress concentration at the bore and keyways will be performed each 40 month period during refueling or maintenance shutdowns coinciding with the service inspection schedule as required by Section XI of the ASME Code.

b. Visual examination of all exposed surfaces will be performed and a surface examination of the bore and keyway surfaces will be performed whenever the flywheels are removed for maintenance purposes, but not more frequently than once each 10 year interval.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.11 At least one reactor vessel head vent path consisting of two valves in series powered from emergency busses shall be OPERABLE and closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the above reactor vessel head vent path inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.11 Each reactor vessel head vent path shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying all manual isolation valves in each vent path are locked in the open position,

b. Cycling each valve in the vent path through at least one complete cycle of full travel from the control room during COLD SHUTDOWN or REFUELING, and

c. Verifying flow through the reactor vessel head vent paths during venting operations at COLD SHUTDOWN or REFUELING.

SPECIFIC ACTIVITY (Continued)

take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomenon. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. LCO 3.4.9.1, "Pressure/Temperature Limits," limits the pressure and temperature changes during RCS heatup and cooldown to within the design assumptions and the stress limits for cyclic operation.

The PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) contains pressure and temperature (P/T) limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

LCO 3.4.9.1 establishes operating limits that provide a margin to non-ductile failure of the reactor vessel and piping of the Reactor Coolant Pressure Boundary (RCPB). The vessel is the component most subject to non-ductile failure, and the LCO limits apply to the entire RCS, except the pressurizer, which has different design characteristics and operating functions.

10 CFR Part 50, Appendix G, requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Appendix G of 10 CFR Part 50 requires an adequate margin to non-ductile failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section XI, Appendix G.

The neutron embrittlement effect on the material toughness is reflected by increasing the Nil Ductility Reference Temperature (RT_{NDT}) as exposure to neutron fluence increases.

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 and Appendix H of 10 CFR Part 50. The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99, Revision 2.

PRESSURE/TEMPERATURE LIMITS (Continued)

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls during heatup and cooldown, respectively.

The criticality limit curve includes the 10 CFR Part 50, Appendix G, requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.1.1.4, "Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in non-ductile failure of the RCPB, possibly leading to a nonisolable leak or loss-of-coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E, provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

OVERPRESSURE PROTECTION SYSTEMS

The Low Temperature Overpressure Protection (LTOP) System controls the RCS pressure at low temperatures so the integrity of the RCPB is not compromised by violating the P/T limits of 10 CFR Part 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the pressurizer Power-Operated Relief Valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the 10 CFR Part 50, Appendix G, requirements during the MODES in which the LTOP system is necessary.

The reactor vessel material is less ductile at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and the material becomes less resistant to stress at low temperatures. RCS pressure, therefore, is maintained low at low temperatures and is increased only within the limits specified in the PTLR.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only during shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause non-ductile failure of the reactor vessel. LCO 3.4.9.1, "Pressure/Temperature Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

PRESSURE/TEMPERATURE LIMITS (Continued)

LCO 3.4.9.3, "Overpressure Protection Systems," provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all Safety Injection (SI) pumps and all but one charging pump (a centrifugal charging pump) incapable of injection into the RCS and isolation of the SI accumulators. The pressure relief capacity requires either two redundant RCS relief valves, or a depressurized RCS and an RCS vent of sufficient size. One RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the SI actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one centrifugal charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

The LTOP System for pressure relief consists of two PORVs with reduced lift settings, or two Residual Heat Removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve, or a depressurized RCS and a RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to prevent overpressurization for the required coolant input capability.

PORV Requirements

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The LTOP actuation logic monitors both RCS temperature and RCS pressure and determines when a condition is approaching the PTLR limits. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

The PTLR presents the PORV setpoints for the LTOP system. The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient. Having the setpoints of both valves within the limits in the PTLR ensures that the 10 CFR Part 50, Appendix G, limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the subsequent relief will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

PRESSURE/TEMPERATURE LIMITS (Continued)

RHR Suction Relief Valve Requirements

During the LTOP MODES, the RHR System is operated for decay heat removal and low pressure letdown control. Therefore, the RHR suction isolation valves are open in the piping from the RCS hot legs to the inlets of the RHR pumps. While these valves are open, the RHR suction relief valves are exposed to the RCS and are able to relieve pressure transients in the RCS.

The RHR suction isolation valves must be open to make the RHR suction relief valves OPERABLE for RCS overpressure mitigation. The RHR suction relief valves are spring loaded, bellows-type relief valves with pressure tolerances and accumulation limits established by Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for Class 2 relief valves.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths.

For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve; removing a PORV's internals, and disabling its block valve in the open position; or similarly establishing any comparable vent. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

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 Part 50.55a(g) except where specific written relief has been granted by the Commission purusant to 10 CFR Part 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.4.11 REACTOR VESSEL HEAD VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of a reactor vessel head vent path ensures the capability exists to perform this function.

3/4.4.11 REACTOR VESSEL HEAD VENTS (Continued)

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single vent valve power supply or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System Vent Systems are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

REPORTING REQUIREMENTS (Continued)

OPERATING LIMITS REPORT (Continued)

9. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," dated August 1985 (Westinghouse Proprietary). (Methodology for Specification: Axial Flux Difference, Heat Flux Hot Channel Factor, and Nuclear Enthalpy Rise Hot Channel Factor).

10. ComEd letter from D. Saccomando to the Office of Nuclear Reactor Regulation dated December 21, 1994, transmitting an attachment that documents applicable sections of WCAP-11992/11993 and ComEd application of the UET methodology addressed in "Additional Information Regarding Application for Amendment to Facility Operating Licenses-Reactivity Controls Systems."

The operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

The OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

REACTOR COOLANT SYSTEM (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

6.9.1.10 RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing, as well as heatup and cooldown rates and pressurizer power-operated relief valve lift settings, shall be established and documented in the PTLR for the following:

1) LCO 3.4.9.1, "Pressure/Temperature Limits," and
2) LCO 3.4.9.3, "Overpressure Protection Systems."

The analytical methods used to determine the RCS pressure and temperature limits shall be those reviewed and approved by the NRC, specifically those described in NRC letter dated January 21, 1998, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report."

The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. NPF-37,

AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. NPF-66,

AMENDMENT NO. 89 TO FACILITY OPERATING LICENSE NO. NPF-72,

AND AMENDMENT NO. 89 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 May 21, 1997, Commonwealth Edison Company (ComEd, the licensee) proposed changes to the technical specifications (TS) for Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2. The requested changes are the relocation of the pressure temperature (P/T) limit curves and low-temperature overpressure protection (LTOP) system limits to the Pressure and Temperature Limits Report (PTLR) and the referencing of that report in the affected limiting conditions for operation and bases. The proposed changes also include the addition of the PTLR to the definitions section of the TS and the addition of a new section to the reporting requirements in the administrative controls section of the TS delineating the necessary reports. Guidance on the proposed changes was developed by the Nuclear Regulatory Commission (NRC) on the basis of a proposal by the owners groups during the development of the improved standard technical specifications (STS). This guidance was provided to all power reactor licensees and applicants by Generic Letter (GL) 96-03, dated January 31, 1996.

ComEd supplemented the May 21, 1997, submittal by letters dated November 18, 1997, December 3, 1997, January 8, 1998 and January 13, 1998. The January 8, 1998 and January 13, 1998, submittals provided additional clarifying information that did not change the initial proposed no significant hazards consideration determination.

In addition, as part of the methodology for developing the P/T limit curves, ComEd submitted a request to integrate the reactor vessel surveillance programs for Byron, Units 1 and 2, and Braidwood, Units 1 and 2, by letter of May 6, 1997, as supplemented in the December 3, 1997, letter.

2.0 BACKGROUND

Section 182a of the Atomic Energy Act (the Act) requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory

9802060192 980123
PDR ADOCK 05000454
P PDR

requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories: (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls, and states also that the Commission may include such additional TS as it finds to be appropriate. However, the regulation does not specify the particular requirements to be included in a plant's TS.

10 CFR 50.36 identifies four criteria to be used in determining whether a particular matter is required to be included in the TS, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. As a result, existing TS requirements which fall within or satisfy any of these criteria must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents.

3.0 EVALUATION

All components of the reactor coolant system (RCS) are designed to withstand the effects of cyclic loads resulting from system pressure and temperature changes. These loads are introduced by heatup and cooldown operations, power transients, and reactor trips. In accordance with Appendix G to 10 CFR Part 50, TS limit the pressure and temperature changes during RCS heatup and cooldown within the design assumptions and the stress limits for cyclic operation. These limits are defined by P/T limit curves for heatup, cooldown, LTOP, and inservice leak and hydrostatic testing. Each curve defines an acceptable region for normal operation. The curves are used for operational guidance during heatup and cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The licensee used the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N-514 or the 1996 Addenda to the ASME Code, and requested an exemption from Appendix G by justifying why Appendix G can not be met and demonstrating that a specific exemption was warranted under 10 CFR 50.12. The exemption to permit use of Code Case N-514 was granted on November 29, 1996, for Byron, Units 1 and 2; July 13, 1995, for Braidwood, Unit 1; and December 12, 1997, for Braidwood, Unit 2. The plants were granted an exemption to permit use of the 1996 Addenda to the ASME Code on January 16, 1998.

The licensee integrated the reactor vessel weld metal surveillance program for Byron, Units 1 and 2, and for Braidwood, Units 1 and 2, pursuant to 10 CFR Part 50, Appendix H, Section III.C. The data integration is consistent with 10 CFR 50.61 (c)(2), which indicates that licensees should consider information from related surveillance programs in assessing the embrittlement of their vessel. The limitations on the data integration are specificied in the PTLR methodology that was approved by the NRC and documented by letter dated January 16, 1998.

The LTOP system controls RCS pressure at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating 10 CFR Part 50, Appendix G. The LTOP system for pressure relief consists of power-operated relief valves (PORVs), residual heat removal (RHR) suction relief valves, or a combination of both. The LTOP system limits consist of PORV and RHR suction relief valve setpoints. The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs and, therefore, are still addressed in the TS. The LTOP system is reevaluated each time the P/T limit curves are revised to ensure that it meets its intended function.

The licensee-proposed changes to the TS are in accordance with the guidance in GL 96-03, as follows:

(1) The definitions section of the TS was modified to include a definition of the PTLR to which the figures, values, and parameters for P/T and LTOP system limits will be relocated on a unit-specific basis in accordance with a methodology approved by the NRC that maintains the acceptance limits and the limits of the safety analysis. As noted in the definition, plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the P/T and LTOP system limits with a reference to the PTLR that provides these limits:

LCO 3.4.9.1, "Pressure/Temperature Limits," and
LCO 3.4.9.3, "Overpressure Protection Systems."

(3) Specification 6.9.1.10, "Reactor Coolant System Pressure and Temperature Limits Report," was added to the reporting requirements of the administrative controls section of the TS. This specification requires that the PTLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the regional administrator and resident inspector. The report provides the explanations, figures, values, and parameters of the P/T and LTOP system limits for the applicable effective period. Furthermore, this specification requires that the figures, values, and parameters be established using the methodology approved by the NRC for this purpose in the NRC letter approving a plant-specific methodology as referenced in the TS and be consistent with all the applicable acceptance limits and the limits of the safety analysis.

Finally, the specification requires that all changes in values of these limits be documented in the PTLR each effective period and submitted upon issuance to the NRC.

Relocation of the P/T curves and LTOP setpoints does not eliminate the requirement to operate in accordance with the limits specified in Appendix G to 10 CFR Part 50. The requirement to operate within the limits in the PTLR is specified in, and controlled by, the TS. Only the figures, values, and parameters associated with the P/T limits and LTOP setpoints are to be relocated to the PTLR. In order for the curves and setpoints to be relocated to a PTLR, a methodology for their development must be reviewed and approved in advance by the NRC. The methodology to be approved by the NRC is to be developed in accordance with GL 96-03. This GL provides guidance regarding referencing the methodology and development of the PTLR including, but not limited to, the requirements of Appendix G to 10 CFR Part 50. Since the methodology is referenced in the TS, changes to the methodology must be approved by the NRC. Further, when

changes are made to the figures, values, and parameters contained in the PTLR, the PTLR is to be updated and submitted to the NRC upon issuance.

On this basis, the NRC staff concludes that the licensee provided an acceptable means of establishing and maintaining the detailed values of the P/T limit curves and LTOP system limits. Further, because plant operation continues to be limited in accordance with the requirements of Appendix G to 10 CFR Part 50 and the P/T and LTOP system limits in the TS will be established using a methodology approved by the NRC, these changes will not impact plant safety.

The staff also concludes that the above-relocated requirements relating to the P/T limits and LTOP system limits are not required to be in the TS under 10 CFR 50.36, and are not required to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety. Accordingly, the staff concludes that the proposed changes are acceptable and that these requirements may be relocated from the TS to the PTLR.

A detailed discussion of the staff's basis for acceptance of the licensee's proposed methodology is provided in the attached letter from R. Capra to O. Kingsley, "Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance of Referencing of Pressure Temperature Limits Report (TAC Nos. M98799, M98800, M98801 and M98802), dated January 21, 1998.

During review of the proposed amendment, the staff identified the need for changes to TS 6.9.1.10. The changes which were made to improve the clarity of TS 6.9.1.10, were discussed with the licensee during a telephone conversation on January 22, 1998. The licensee agreed with the staff's suggested changes.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 and change surveillance requirements. The NRC 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 previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (62 FR 66394). Accordingly, the 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 the amendments.

6.0 CONCLUSION

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

Attachment: As stated

Principal Contributor: S. Bailey

Date: January 23, 1998

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

January 21, 1998

Mr. Oliver D. Kingsley, President
Attn.: Regulatory Services
Nuclear Generation Group
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

SUBJECT: BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2, ACCEPTANCE FOR REFERENCING OF PRESSURE TEMPERATURE LIMITS REPORT (TAC NUMBERS M98799, M98800, M98801, AND M98802)

Dear Mr. Kingsley:

REFERENCES:

1. Letter from R. R. Assa, NRC, to D. L. Farrar, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60 - Braidwood Station, Unit 1," July 13, 1995.

2. Letter from C. I. Grimes, NRC, to R. A. Newton, Westinghouse Electric Corporation, "Acceptance for Referencing of Topical Report WCAP-14040, Revision 1, 'Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,'" October 16, 1995. (Also known as WCAP-14040-NP-A).

3. Letter from G. F. Dick, NRC, to I. M. Johnson, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60, 'Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation' - Byron Station, Units 1 and 2," November 29, 1996.

4. Letter from J. B. Hosmer, Commonwealth Edison Company, to NRC Document Control Desk, "Reactor Vessel Integrated Surveillance Program 10 CFR 50, Appendix H, Section III. C," May 6, 1997. (WCAP-14824, Revision 1 is Attached).

5. Letter from J. B. Hosmer, Commonwealth Edison Company, to NRC Document Control Desk, "Application for Amendment to Appendix A, Technical Specifications, for Facility Operating Licenses Relocation of Pressure and Temperature Limits," May 21, 1997.

6. Letter from J. B. Hosmer, Commonwealth Edison Company, to NRC Document Control Desk, "Supplement to the Application for Amendment to Appendix A, Technical Specifications, for Facility Operating Licenses Relocation of Pressure and Temperature Limits," November 18, 1997. (WCAP 14940 and Errata to WCAP-14940 and WCAP-14970 are Attached).

Contact: Maggalean W. Weston, (301) 415-3151

ATTACHMENT

7. Letter from J. B. Hosmer, Commonwealth Edison Company, to NRC Document Control Desk, "Supplemental Information Pertaining to Byron & Braidwood's Reactor Vessel Integrated Surveillance Program," December 3, 1997. (WCAP-14824, Revision 2, and Erratum to WCAP-14824, Revision 2 are Attached).

8. Letter from G. F. Dick, NRC, to I. M. Johnson, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60 - Braidwood Nuclear Station, Unit 2," December 12, 1997.

9. Letter from H. G. Stanley, Commonwealth Edison Company, to NRC Document Control Desk, "Supplemental Information Pertaining to Technical Specification Amendment Regarding Pressure Temperature Curves - Byron and Braidwood Nuclear Power Stations," January 8, 1998. (Errata to WCAP-14824 Revision 2, WCAP-14940 and WCAP-14970 are Attached).

10. Letter from H. G. Stanley, Commonwealth Edison Company, to NRC Document Control Desk, "Supplemental Information Pertaining to Technical Specification Amendment Regarding Pressure Temperature Curves - Byron and Braidwood Nuclear Power Stations," January 13, 1998.

11. Letter from G. F. Dick, NRC, to O. D. Kingsley, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60 - Byron, Units 1 and 2, and Braidwood, Units 1 and 2," January 16, 1998.

12. Letter from R. A. Capra, NRC, to O. D. Kingsley, Commonwealth Edison Company, "Integration of Reactor Pressure Vessel Surveillance Program for Byron and Braidwood, Units 1 and 2," January 16, 1998.

We have completed our review of the pressure temperature (P/T) limit curves and low temperature overpressure protection (LTOP) system limits methodology and the pressure temperature limits report (PTLR) (as referenced above), submitted by the Commonwealth Edison Company (ComEd). We find the methodology to be acceptable for referencing in the administrative controls section of the Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2, technical specifications (TS) to the extent specified and under the limitations delineated in your submittals and the associated NRC safety evaluation, which is enclosed. The safety evaluation defines the basis for acceptance of the submittals. Our acceptance applies only to the matters described in the submittals.

The NRC reviewed ComEd's method for integrating the data from the reactor pressure vessel surveillance capsules as a part of the methodology for developing the P/T limit curves. While the staff finds the method acceptable, as stated in the safety evaluation, the NRC notes that ComEd should (1) re-evaluate the appropriate method for determining the best-estimate chemical composition as additional chemical composition data become available, (2) assess the impact of the assumption that the vessel weld has the same $RT_{NDT(U)}$ value as determined from the surveillance weld in future revisions to the PTLR and/or when additional surveillance data

become available, and (3) address the method for assessing the credibility of the surveillance capsule data, including the method for accounting for irradiation environment and chemical composition differences, in future revisions to the PTLR.

The methodology for review relating to the P/T limit curves and the LTOP system limits was provided in the references listed above. Reference 2 included WCAP-14040-NP-A which provided parts of the methodology used for determining the acceptance of the Byron and Braidwood methodology.

The methodology in WCAP-14040-NP-A, along with supplements provided by ComEd will be used to calculate future changes to the P/T limit curves and LTOP system limits. ComEd may generate new P/T limit curves and LTOP system limits in accordance with this methodology without prior approval of the staff. However, changes to the methodology must first be reviewed and approved by the staff. System limits may be subject to audit by the staff through inspections as necessary.

We do not intend to repeat our review of the matters described in the submittals if the submittals appear as references in other license applications relating to your plants, except to ensure that the material in the submittals is still applicable to your plants as indicated in the conclusion section of the safety evaluation.

Should our criteria or regulations change so that our conclusions as to the acceptability of the methodology is invalidated, licensees referencing these documents will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the documents without revision of their respective documentation.

Sincerely,

Robert A. Capra

Robert. A. Capra, Director
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. STN 50-454, STN 50-455
STN 50-456 and STN 50-457

Enclosure: Safety Evaluation

cc w/encl: See next page

O. Kingsley
Commonwealth Edison Company

cc:

Mr. William P. Poirier, Director
Westinghouse Electric Corporation
Energy Systems Business Unit
Post Office Box 355, Bay 236 W.
Pittsburgh, Pennsylvania 15230

Joseph Gallo
Gallo & Ross
1250 Eye St., N.W., Suite 302
Washington, DC 20005

Michael I. Miller, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Howard A. Learner
Environmental law and Policy
Center of the Midwest
203 N. LaSalle St. Suite 1390
Chicago, Illinois 60601

U.S. Nuclear Regulatory Commission
Byron Resident Inspectors Office
4448 N. German Church Road
Byron, Illinois 61010-9750

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

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

Chairman, Ogle County Board
Post Office Box 357
Oregon, Illinois 61061

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

George L. Edgar
Morgan, Lewis and Bochius
1800 M Street, N.W.
Washington, DC 20036

Byron/Braidwood Stations
January 21, 1998

Attorney General
500 S. Second Street
Springfield, Illinois 62701

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

Commonwealth Edison Company
Byron Station Manager
4450 N. German Church Road
Byron, Illinois 61010-9794

Commonwealth Edison Company
Site Vice President - Byron
4450 N. German Church Road
Byron, Illinois 61010-9794

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

Mr. Ron Stephens
Illinois Emergency Services
and Disaster Agency
110 E. Adams Street
Springfield, Illinois 62706

Chairman
Will County Board of Supervisors
Will County Board Courthouse
Joliet, Illinois 60434

Commonwealth Edison Company
Braidwood Station Manager
RR 1, Box 84
Braceville, Illinois 60407

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

Document Control Desk-Licensing
Commonwealth Edison Company
1400 Opus Place, Suite 400
Downers Grove, Illinois 60515

Commonwealth Edison Company
Site Vice President - Braidwood
RR 1, Box 84
Braceville, IL 60407

Mr. Michael J. Wallace
Nuclear Services Senior Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, IL 60515

Mr. Gene H. Stanley
PWR's Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, IL 60515

Mr. Steve Perry
BWR's Vice President
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 900
Downers Grove, IL 60515

Mr. Dennis Farrar
Regulatory Services Manager
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

Ms. Irene Johnson, Licensing Director
Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, IL 60515

Commonwealth Edison Company
Reg. Assurance Supervisor - Braidwood
RR 1, Box 79
Braceville, Illinois 60407

Commonwealth Edison Company
Reg. Assurance Supervisor - Byron
4450 N. German Church Road
Byron, Illinois 61010-9794

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REVIEW OF THE PRESSURE TEMPERATURE LIMITS REPORT (PTLR) AND METHODOLOGY FOR THE RELOCATION OF THE REACTOR COOLANT SYSTEM (RCS) PRESSURE-TEMPERATURE LIMIT CURVES AND LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) SYSTEM LIMITS

RELATED TO FACILITY OPERATING LICENSES NPF-37, NPF-66, NPF-72 AND NPF-77

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNITS 1 AND 2, AND BRAIDWOOD STATION, UNITS 1 AND 2

DOCKET NUMBERS 50-454, 50-455, 50-456, AND 50-457

1.0 INTRODUCTION

By letter dated May 21, 1997 (Reference 5), and supplemented by letters dated November 18, 1997 (Reference 6), December 3, 1997 (Reference 7), January 8, 1998 (Reference 9), and January 13, 1998 (Reference 10), Commonwealth Edison Company (ComEd), requested changes to the technical specifications (TS) for Byron Units 1 and 2, and Braidwood Units 1 and 2. The requested changes included (1) developing new reactor coolant system (RCS) pressure temperature (P/T) limit curves and low temperature overpressure protection (LTOP) system limits, (2) relocating the P/T limit curves and LTOP system limits from the TS to a licensee controlled document identified as a Pressure Temperature Limits Report (PTLR), and (3) changing the affected limiting conditions for operation and bases accordingly. These changes are made in accordance with Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996.

In addition, ComEd also requested a change to relocate the surveillance capsule withdrawal schedule from the TS to the PTLR. Relocation of the reactor vessel surveillance capsule withdrawal schedule is made in accordance with GL 91-01, "Removal of the Schedule for the Withdrawal of Reactor Vessel Material Specimens from Technical Specifications," except that the schedule is relocated to the PTLR rather than the UFSAR. Changes to this schedule continue to be controlled by the requirements of Appendix H to 10 CFR Part 50.

The licensee's request to relocate the P/T limit curves and LTOP system limits to the PTLR was submitted consistent with the guidance provided in GL 96-03 and WCAP-14040-NP-A (Reference 2), with three exceptions. These exceptions, discussed in Attachment E of the licensee's May 21, 1997, submittal, involved (1) the computer program used for determination of the LTOP setpoints, (2) the neutron transport cross-section library and dosimeter reaction cross sections used in determining the fluences, and (3) the version of the ASME Code used in determining the P/T limit curves and LTOP system limits . Exceptions 1 and 2 are addressed elsewhere in this evaluation. Exception 3 was addressed, in part, in the evaluation authorizing the use of the methodology in the ASME Section XI, Appendix G, 1996 Addenda (Reference 11).

9802050202 980123
PDR ADOCK 05000454
P PDR

The Commission granted Byron and Braidwood exemptions (References 1, 3, and 8) from the requirements of 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation," that allowed the plants to use the methodology in ASME Code Case N-514. Byron and Braidwood were also granted exemptions (Reference 11), permitting the use of the methodology described in the 1996 ASME Code, Section XI, Appendix G, Article G-2000. In addition, the staff approved (Reference 12), the integration of the Byron 1 and 2 and Braidwood 1 and 2 weld metal surveillance programs.

With regard to the LTOP system limits, the Code Case and the 1996 Addenda provide essentially the same guidance, in that they both recommend that LTOP systems limit the pressure in the vessel to 110% of the P/T limits and allow the use of enable temperatures less than 200°F or a coolant temperature corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50°F, whichever is greater.

2.0 BACKGROUND

2.1 Neutron Fluence

The fluence evaluation which is the basis for the current P/T curves was performed when the first four surveillance capsules were removed and evaluated at the end of the first cycle for each plant. However, at the time these evaluations were performed, ENDF/B-IV based cross sections were used. Since that time the staff has identified a number of nonconservative values in the Iron inelastic cross sections. The cross sections now recommended by the staff are based on the ENDF/B-VI cross section file.

The licensee proposed to move the P/T limit curves to the PTLR before an integrated fluence reevaluation is performed based on the ENDF/B-VI cross sections. This is scheduled to occur after the removal of the next set of surveillance capsules. This reevaluation will change the manner in which the materials are utilized, and, therefore will change the PTLR methodology. Because of the changes, the methodology will then have to be submitted to the NRC for prior review and approval. The licensee stated that the maximum operating time used to generate the P/T limit curves will be at most 85.8% of the current licensing basis time. Therefore, the revised operating time used to generate the curves will be no greater than 85.8% of the currently approved maximum value. This evaluation establishes a conservative fluence estimation. The nonconservatism of the ENDF/B-IV file is about 20%. The conservatisms are 14.2%, due to the shorter operating time; 6%, due to the initial adjustments to the measured data; and 5%, due to the effect of the low leakage loadings which have been practiced in all units since the second cycle.

2.2 Pressure Temperature Limits

The methodologies for assessing P/T limits and reactor pressure vessel (RPV) surveillance programs are discussed, in part, in the following documents: (1) 10 CFR Part 50, "Appendix G - Fracture Toughness Requirements"; (2) 10 CFR Part 50, "Appendix H - Reactor Vessel Material Surveillance Program Requirements"; (3) 10 CFR 50.60 - "Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation"; (4) 10 CFR 50.61 - "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"; and (5) Regulatory Guide 1.99, Revision 2 - "Radiation Embrittlement of Reactor Vessel Materials." The terms and methods used throughout this evaluation are discussed in detail in these sources.

In 1997, ComEd submitted several requests to modify its methodology for determining the P/T limit curves for Byron 1 and 2 and Braidwood 1 and 2. One of the changes involved using a later version of the ASME Boiler and Pressure Vessel Code than that listed in 10 CFR 50.55a for determination of the P/T limit curves (i.e.,the licensee requested to use the methodology specified in the 1996 ASME Code, Section XI, Appendix G, Article G-2000 rather than the 1989 ASME Code). This change primarily altered the methodology used in determining the applied stress intensity due to thermal stresses. The other change involved integrating the weld metal surveillance programs for Byron Units 1 and 2 and for Braidwood Units 1 and 2.

2.3 Low Temperature Overpressure Protection System

The LTOP system mitigates overpressure transients at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating 10 CFR Part 50, Appendix G. The LTOP systems at Byron and Braidwood use combinations of pressurizer power operated relief valves (PORVs) and residual heat removal (RHR) suction relief valves to accomplish this function. The PORV portion of the system is manually enabled. When enabled, the system continuously monitors RCS temperature and pressure conditions. An auctioneered system temperature is continuously converted to an allowable pressure and then compared to the actual RCS pressure. The system logic first annunciates a main control board alarm whenever the measured pressure reaches a predetermined setpoint, thereby indicating a pressure transient is occurring. On a further increase in measured pressure, an actuation signal opens the PORVs in order to prevent pressure-temperature conditions from exceeding allowable limits. The RHR suction relief valves have a constant setpoint and are available for low temperature overpressure protection whenever the corresponding RHR train is placed in service.

The design basis of the LTOP system considers both mass-addition and heat-addition transients during water solid RCS conditions. The mass-addition analyses account for the injection from one charging pump. The heat-addition analyses account for heat input from the secondary sides of all steam generators (SGs) into the RCS, upon starting a single reactor coolant pump (RCP). The heat-addition transient analyses assume the secondary side temperatures of the SGs are 50°F higher than the RCS temperature. The Byron and Braidwood proposed LTOP enable temperatures and actuation setpoints were established using the methodology presented in WCAP-14040-NP-A, in combination with ASME Code Case N-514 and ASME Section XI, Appendix G, 1996 Addenda.

3.0 EVALUATIONS

3.1 Neutron Fluence Evaluation

The current capsule withdrawal schedule is as follows:

Byron	Unit 1:	Capsule W,	Cycle B1R08,	November,	1997
Byron	Unit 2:	Capsule X,	Cycle B2R07,	April,	1998
Braidwood	Unit 1:	Capsule W,	Cycle A1R07,	September,	1997
Braidwood	Unit 2:	Capsule W,	Cycle A2R07,	May,	1999

For the total Iron thickness of about 5.2 inches (2.5 inches for the core barrel and 2.7 inches for the neutron pads) the staff conservatively estimates that the underprediction of the vessel fluence due to the Iron cross sections in ENDF/B-IV is about 20%.

The projected operating time for Byron Unit 1 is 10.3 effective full power years (EFPYs) to the end of refueling cycle 9 at Byron Unit 1 (B1R09). Given that the currently approved license value is 12 EFPYs, operation would continue for 10.3/12 = 85.8% of the approved time. This is the longest operating time at all four units and is conservatively assumed to be applicable for all units. Byron 2 is projected to operate for 9.9 EFPYs until B2R08. It is currently approved for operation for 12 EFPYs, so this modification to the P/T curves will apply for 9.9/12 = 82.5% of the approved time.

Braidwood Unit 1 is projected to operate for 9.56 EFPYs until A1R08. It is currently approved for 16 EFPYs, so this modification to the P/T curves will apply for 9.56/16 = 59.8% of the approved time. Braidwood Unit 2 is projected to operate for 10.18 EFPYs until A2R08. It is currently approved for 12 EFPYs, so this modification to the P/T curves will apply for 10.18/12 = 84.8%.

In addition, the licensee stated that, at the time of the current fluence calculation, the final values were increased by about 6% of the measured value from the capsule dosimetry. Therefore, this is a conservatism with respect to the calculated value. There is an estimated 5% conservatism due to the low leakage loadings practiced from cycle 2 to the present.

Given that the fluence is increasing linearly with respect to the number of EFPYs, the staff has determined that for Byron Unit 1 the nonconservatism amounts to about 20%. However, the conservatisms in the determination of the fluence amount to about 14.2 + 5 + 6 = 25.2%. These conservatisms are, therefore, larger that the nonconservatism due to the use of the ENDF/B-IV cross section file. The staff, therefore, believes that the fluence values the licensee proposes to use to generate the P/T curves are acceptable. The remaining units are even more conservative than Byron Unit 1, thus, they are also acceptable.

3.2 Pressure Temperature Limits Evaluation

Based on the information provided by the licensee, the methods used by the licensee in determining the vessel material data conform, in general, to the methodology approved by the NRC in WCAP-14040-NP-A which endorses Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." Although the methodology used by the licensee is acceptable and consistent with that previously approved (WCAP-14040-NP-A), there are some details of the methodology (not specifically addressed in WCAP-14040-NP-A) that require additional comment. These details include (1) the method used in determining the best-estimate chemistry for welds, (2) the appropriate unirradiated reference temperature ($RT_{NDT(U)}$) value used in determining the adjusted reference temperature (ART), and (3) the method for assessing integrated surveillance data. These areas are discussed below. In addition, the NRC's evaluation regarding the adequacy of the ARTs and the P/T limit curves for each unit is provided below, as are the details regarding implementation of the 1996 methodology that was addressed in reviewing the acceptability of the P/T limits.

3.2.1 Weld-Best Estimate Chemistry

The best-estimate chemical composition (copper and nickel) for a heat of weld metal can be determined by several methods. Three of the more frequently used methods are the simple average, the mean-of-the-means, and the coil weighted average method (copper only). In the simple average method, all of the chemical composition data for that heat of material are averaged regardless of the source. (Sources of chemical composition data include weld metal qualification tests, a plant's surveillance program, and nozzle dropout analyses.) In the mean-of-the-means approach, the mean value for each source of data for that particular heat of material is averaged to determine the best-estimate chemical composition for the heat. In the coil

weighted average method, the mean value for each source of data is weighted by the number of coils of wire used in the fabrication of the weld. The weighted average (mean) for each group is then averaged to determine the best-estimate chemical composition for the heat of material.

Selection of the appropriate method to use requires significant technical judgment. For example, the simple average method may be adversely influenced by numerous chemical analyses from one source of data. The mean-of-the-means approach, however, avoids this by placing equal weight on each source (which can also be a disadvantage since it gives a source of data with 1 chemistry sample the same weight as a source of data with 20 chemistry samples).

In addition, when a mean-of-the-means or coil-weighted average approach is used in determining the best-estimate chemistry, the way in which the chemistry data are grouped (in particular, those from weld qualification tests) can have a significant impact on the results. That is, the resultant best estimate value can vary depending on whether the chemistry data are determined to be from "one weld" or from multiple welds. If weld qualification specimens were fabricated by the same manufacturer, within a short time span, using similar welding input parameters, and using the same coil of weld consumables, the staff's recommendation is that all chemistry samples from that weld should be considered as "one weld" for the purposes of best-estimate chemistry determination. If information is not available to confirm the aforementioned details, but sufficient evidence exists to reasonably assume the details are the same, the best-estimate chemistry should be evaluated both by assuming the data came from "one weld" and by assuming that the data came from an appropriate number of "multiple welds."

The licensee concluded that the mean-of-the-means approach is the most appropriate method since it eliminates the inappropriate weighting effect which results from numerous analyses from a particular weld block (i.e., source). The licensee further concluded that a coil weighted average approach is not a fundamentally sound basis for evaluating weld chemistry because of the lack of documentation of coil changes or intra-coil splices that may have occurred or been present during production of the welds.

Given that the licensee concluded that the mean-of-the-means approach was the most appropriate method, the staff evaluated the method used by the licensee to group the weld qualification test data. The licensee chose to group the weld qualification data as coming from multiple sources since information explicitly linking the data to other data was not available. The two heats of weld metal potentially affected by this issue are heats 442002 and 442011. With this approach, for heat 442002, the licensee determined that the mean-of-the-means value is 0.053% copper and 0.621% nickel. If the weld qualification test for a given weld is assumed to come from the same block, the mean-of-the-means value would be 0.059% copper and 0.628% nickel. The staff used 0.059% copper and 0.628% nickel as the best-estimate chemical composition for heat 442002 and determined that the surveillance data were credible. Using the ratio procedure and the surveillance data, the chemistry factor was determined by the staff to be less (which is less conservative) than the value used by the licensee. For heat 442011, the licensee determined that the mean-of-the-means value is 0.032% copper and 0.666% nickel. If the weld qualification test for a given weld is assumed to come from the same block, the mean-of-the-means value would be 0.033% copper and 0.667% nickel. The staff concluded that there were no appreciable differences in the ART using either of these chemical compositions for heat 442002.

Only heats 442002 and 442011 were impacted by the method (simple average, mean-of-the-means, coil weighted average) chosen to determine the best-estimate chemical composition. For heat 442002, the mean-of-the-means approach, which the licensee used, provides the most

conservative estimate of the chemical composition for the heat. (Copper and nickel are 0.029% and 0.680%, respectively, for the simple average method. Copper and nickel are 0.059% and 0.628%, respectively, for the mean-of-the-means approach). On the other hand, for heat 442011, the simple average method yields a slightly more conservative chemical composition. (Copper and nickel are 0.033% and 0.688%, respectively, for the simple average method. Copper and nickel are 0.032% and 0.667%, respectively, for the mean-of-the-means approach).

Given the differences in the estimates by the three methods and the number of samples for each source of data, the staff concludes that the values used by the licensee are acceptable. However, as additional chemical composition data become available, the licensee should re-evaluate the appropriate method for determining the best-estimate chemical composition and the appropriate method for grouping the data if the mean-of-the-means or coil weighted average approach are used.

3.2.3 Unirradiated Reference Temperature

The unirradiated reference temperature, $RT_{NDT(U)}$, is used in the determination of the ART. As discussed during a public meeting on November 12, 1997 (see meeting summary dated November 19, 1997, "Meeting Summary for November 12, 1997 Meeting with Owners Group Representatives and NEI Regarding Review of Responses to Generic Letter 92-01, Revision 1, Supplement 1 Responses"), the staff recognized that in some instances there are significant differences in the $RT_{NDT(U)}$ values used by licensees for the same heat of material. The NRC also indicated that this was a potential long-term issue.

For Byron and Braidwood Units 1 and 2, the $RT_{NDT(U)}$ values used in the evaluation differ for several heats of material as shown in Table 1. The licensee believes that the differing $RT_{NDT(U)}$ values may be explained on the basis of the different flux lots used in the fabrication of the weld.

TABLE 1: DIFFERENCES IN $RT_{NDT(U)}$ FOR MATERIALS IN BYRON 1 AND 2

Heat of Material	$RT_{NDT(U)}$			
	Byron 1	Byron 2	Braidwood 1	Braidwood 2
31401	10.0	40.0	-40.0	N/A
442002	-30.0	10.0	N/A	N/A
442011	10.0	40.0	40.0	40.0

As shown in Table 1, heat 31401 has the greatest variability. The staff evaluated this variability and concluded that even if the most limiting $RT_{NDT(U)}$ value (i.e., 40.0 °F) was used in the Byron 1 and Braidwood 1 evaluation of this heat of material, the limiting material would not change and the RT_{PTS} screening criteria would not be exceeded. This calculation was performed using the end-of-license fluence reported in the NRC Reactor Vessel Integrity Database. Similarly, for heat 442011, the limiting material would not change and the RT_{PTS} screening criteria would not be exceeded for Byron 1 if an $RT_{NDT(U)}$ value of 40.0°F was used. In this evaluation, the staff assumed that the surveillance data from Braidwood 1 and 2 heat 442011 could be used in the evaluation of the Byron 1 vessel based on the similarity of the irradiation environments.

For heat 442002, if the limiting $RT_{NDT(U)}$ value (i.e., 10°F) is used in the Byron 1 evaluation of this heat, this would become the limiting material by approximately 10°F but the RT_{PTS} screening criteria would not be exceeded. For Byron 1, the weld in question is identified as WF-336 and was fabricated from weld wire heat 442002 and flux lot 8873. The similar weld at Byron 2 is identified as WF-447 and was fabricated from weld wire heat 442002 and flux lot 8064. The licensee believes that the differing initial $RT_{NDT(U)}$ values may be explained on the basis of the different flux lots used. The staff finds this acceptable; nonetheless, the staff still considers this an issue that may lead to rule changes as noted in the November 12, 1997, meeting. In future revisions to the PTLR, the licensee should assess the impact of its assumption that the vessel weld has the same $RT_{NDT(U)}$ value as determined from the surveillance weld.

3.2.4 Credibility Evaluation for Integrated Surveillance Program

To assess the credibility of surveillance data from welds when the data are from more than one source, the data should be normalized to account for differences in chemical composition and irradiation environment consistent with NRC regulations. The NRC provided guidance on performing these adjustments in the November 12, 1997, public meeting as discussed above. In assessing the credibility of the data, the licensee did not adjust for chemical composition differences (i.e., apply the ratio procedure). The licensee did consider differences in the irradiation temperature between the specimens at Byron 1 and Byron 2 and concluded that the differences were small.

The staff, on the other hand, assessed the credibility of the integrated weld metal surveillance program by normalizing the surveillance data to the average chemical composition and irradiation environment of the surveillance specimens as discussed below. The irradiation environment (i.e., temperature) to which the surveillance weld metal at Byron 1 and 2 were exposed were considered identical to those to which the vessels were exposed (consistent with the licensee's conclusion that the effect is small); therefore, no adjustments to the surveillance data to account for temperature were made in assessing the credibility of the data. To account for chemical composition differences, the staff used the ratio procedure and normalized the data to the average chemical composition of the surveillance welds (i.e., copper = 0.0225; nickel = 0.701). Even though the details between the staff and the licensee's evaluation differed, the net result was the same. That is, the data were determined to be credible. However, in future revisions to the PTLR and/or when additional surveillance data become available, the licensee should address the method for assessing the credibility of the data including the method for accounting for irradiation environment and chemical composition differences. This is consistent with the licensee's statement that "temperature differences and chemistry factor ratios will be re-evaluated at all future scheduled capsule evaluations."

3.2.5 Byron Unit 1

Based on the material provided by the licensee, the staff confirmed that (1) the limiting material is intermediate shell forging 5P-5933 with a 1/4T ART of 70°F and a 3/4T ART of 60°F and (2) the P/T limit curves are appropriate.

3.2.6 Byron Unit 2

Based on the material provided by the licensee, the staff confirmed that (1) the limiting material is circumferential weld, WF-447 (heat 442002) and (2) the P/T limit curves are appropriate. The ARTs calculated by the staff (1/4T ART of 81.9°F and 3/4T ART of 67.6°F) were slightly lower

(i.e., less conservative) than that reported by the licensee (1/4T ART of 87.6°F and 3/4T ART of 71.5°F) as a result of minor differences in chemical composition and in applying the ratio procedure.

3.2.7 Braidwood Unit 1

Based on the material provided by the licensee, the staff confirmed that (1) the limiting material is circumferential weld, WF-562 (heat 442011) and (2) the P/T curves are appropriate. The ARTs calculated by the staff (1/4T ART of 69.7°F and 3/4T ART of 60.6°F) were lower (i.e., less conservative) than that reported by the licensee (1/4T ART of 76.6°F and 3/4T ART of 65.4°F) as a result of using a slightly different chemical composition for the weld. Furthermore, the P/T limit curves are more conservative than would be required using the methodology in WCAP-14040-NP-A. Since the licensee's results are more conservative, the staff concludes that they are appropriate.

3.2.8 Braidwood Unit 2

Based on the material provided by the licensee, the Staff confirmed that (1) the limiting material is circumferential weld, WF-562 (heat 442011) with a 1/4T ART of 66.8°F and a 3/4T ART of 58.1°F and (2) the P/T curves are appropriate.

3.3 Low Pressure Overpressure Protection System Evaluation

3.3.1 LTOP Enable Temperature

The LTOP enable temperature is the temperature below which the LTOP system is required to be operable. The licensee's proposed enable temperature (1) accounts for instrumentation uncertainties associated with the instrumentation used to enable the LTOP system and (2) implement the ASME Code Case/1996 Addenda methodology of using an enable RCS liquid temperature corresponding to the reactor vessel ¼-T metal temperature of RT_{NDT} + 50 or 200°F, whichever is greater. Therefore, the minimum allowed enable temperature was calculated as the larger of either RT_{NDT} + 50°F + E_{inst} + delta-$T_{¼-T}$ or 200°F. In this calculation, E_{inst} refers to instrument error while delta-$T_{¼-T}$ refers to the temperature difference between the reactor coolant and the metal at a distance one-fourth of the vessel wall thickness from the inside surface in the beltline region.

The use of 50°F in the above methodology is consistent with ASME Code Case N-514 and the 1996 Addenda. Accounting for the delta-$T_{¼-T}$ is consistent with Branch Technical Position (BTP) RSB 5-2, which states that the enable temperature is defined as "the water temperature corresponding to the metal temperature...at the beltline location (1/4T or 3/4T) that is controlling in the Appendix G limit calculation." This approach is also consistent with the ASME Code Case and the 1996 Addenda. Accounting for instrument uncertainty ensures that the LTOP system is not enabled at temperatures less conservative than are required by the aforementioned documents. Based on the above discussion the staff finds acceptable the licensee's implementation of the LTOP enable temperature methodology.

The licensee proposed an LTOP enable temperature of ≥ 350°F for all four units. Based on vessel material data, instrumentation uncertainties, and delta-$T_{¼-T}$, the minimum allowed enable temperature for each of the units is as follows:

TABLE 2: ENABLE TEMPERATURES

Plant	RT_{NDT}	E_{inst}	Delta-T_{N-T}	Calculated Enable Temperature	Minimum Allowed Enable Temperature (Greater of Calculated or 200°F)
Braidwood 1	76.6°F	15.0°F	29.254°F	170.9°F	200°F
Braidwood 2	66.9°F	15.0°F	29.254°F	161.2°F	200°F
Byron 1	70.0°F	15.0°F	29.254°F	164.3°F	200°F
Byron 2	87.6°F	15.0°F	29.254°F	181.9°F	200°F

For the above listed RT_{NDT} values, the proposed LTOP enable temperature of ≥ 350°F is conservative with respect to the minimum LTOP enable temperature allowed by WCAP-14040-NP-A, ASME Code Case N-514 and the ASME Section XI, Appendix G, 1996 Addenda . Therefore, the staff finds the licensee's proposed enable temperature acceptable.

3.3.2 LTOP Actuation Setpoint

The ASME Code Case and 1996 Addenda both allow LTOP systems to be designed to limit the peak pressure at the controlling location in the reactor to 110% of the P/T limits. Additionally, since overpressure events most likely occur during isothermal conditions in the RCS, the NRC has approved the use of the steady-state P/T limits for the design of LTOP. WCAP-14040-NP-A provides a methodology for calculating setpoints for LTOP systems that use pressurizer PORVs with variable setpoints. Section 3.2, "COMS Setpoint Determination," of WCAP- 14040-NP-A provides discussions of parameters that need to be considered in determining the LTOP actuation setpoint. A summary of the licensee's LTOP analyses were submitted in the January 8 and 13, 1998, letters. The licensee's analyses and proposed PORV lift setpoints were based on WCAP-14040-NP-A and the P/T limits with 10% relaxation in accordance with ASME Code Case N-514 and the 1996 Addenda. The resulting PORV lift setpoints were provided as Figure 2.3 in each units PTLR.

Westinghouse performed mass-addition and heat-addition LTOP analyses for Byron and Braidwood using the LOFTRAN computer code. Use of the LOFTRAN computer code is consistent with WCAP-14040-NP-A. For the mass addition transients, the RCS was assumed to be water solid. The analyses accounted for injection from a single charging pump and calculated the amount of overshoot that would occur during such a transient. To ensure consistency between the analysis assumptions and the TS, Surveillance Requirement (SR) 4.5.3.2 ensures that all other charging and safety injection (SI) pumps are made incapable of injecting into the RCS while in the LTOP region. This SR, however, uses a temperature of 330°F to make the pumps inoperable instead of the licensee's proposed LTOP enable temperature of 350°F. Using 330°F is acceptable because the minimum required enable temperature (Section 3.1 of this safety evaluation (SE)) is 200°F. A configuration allowing SI pumps to be available when the pressurizer level is less than or equal to 5% is evaluated in Section 3.3.4 of this SE.

Heat-addition transients were also run with the RCS in a water solid condition. For these analyses, the secondary system was assumed to be 50°F higher than the RCS. One RCP was assumed to start and consequent heat addition from all SGs was accounted for. Expansion of the reactor coolant resulted in pressurization of the RCS and actuation of the LTOP system. The resulting overshoot values were determined by these analyses. To ensure consistency between the analyses assumptions and the TS, a note is included in TS 3.4.1.3 to ensure that RCPs are not started unless the secondary side of any SG is less than 50°F higher than the RCS.

For all cases analyzed, the licensee conservatively assumed one PORV failed and, therefore, credited only one PORV for pressure relief. Additionally, the licensee did not credit the RHR suction relief valves for mitigating the pressure transient in the analyses. An evaluation presented in Section 3.3.3 of this SE addresses the substitution of RHR suction relief valves for PORV as currently allowed by TS 3.4.9.3. The licensee evaluated the results of the heat addition and mass-addition cases as a function of temperature and used the more conservative value of overshoot for LTOP setpoint calculations. The licensee accounted for static and dynamic head effects as well as instrumentation uncertainties in the final determination of the LTOP setpoints. The dynamic head effect was divided into two regions. For RCS temperatures above 120°F, the licensee used a pressure drop corresponding to all four RCPs and both RHR pumps running. For RCS temperatures less than or equal to 120°F, the licensee used a pressure drop corresponding to one RCP and both RHR pumps running. Plant administrative procedures prohibit running more than one RCP below 120°F.

The above analyses were performed using the LOFTRAN computer code assuming the original steam generators. For Byron Unit 1 and Braidwood Unit 1, Framatome and ComEd Nuclear Fuel Services performed analyses that verified the Westinghouse calculated setpoints remain valid for the replacement steam generators. These analyses were performed using RELAP5/MOD2-B&W computer code, which is documented in Topical Report BAW-10164P-A. This code was approved by the staff for both loss-of-coolant accidents (LOCA) and non-LOCA applications and is therefore acceptable for use in LTOP analyses.

Based on the above discussion, the staff finds the licensee's implementation of the WCAP-14040-NP-A methodology and the proposed LTOP actuation setpoints as presented in Figures 2.3 of each unit's PTLR acceptable. In addition, the staff finds acceptable the licensee's request to perform LTOP analyses using the NRC approved RELAP5/MOD2-B&W computer code.

3.3.3 RHR Suction Relief Valves

TS 3.4.9.3 requires at least two overpressure protection devices, each consisting of either an RHR suction relief valve or a PORV. This requirement would allow for RHR suction relief valves to be used in place of PORVs for LTOP. WCAP-14040-NP-A does not address the use of the RHR suction relief valves. The staff's SE for the WCAP, dated October 16, 1995, states that licensees who use the WCAP should address this issue in their PTLR submittal. The licensee addressed the use of the RHR suction relief valves in its letter dated January 8, 1998. The licensee's evaluation justified the use of the RHR suction relief valves at its current lift setpoint of 450 psig and appropriately accounted for a setpoint drift of 3% and an accumulation of 10% as recommended by Article NC-7000 of the ASME Boiler and Pressure Vessel Code. The staff reviewed the licensee's evaluation and found it acceptable.

The licensee also included a requirement in the PTLR to evaluate the RHR suction relief valves in a similar manner whenever the P/T limits are revised. The staff finds this consistent with the recommendation in the SE for WCAP-14040-NP-A and, therefore, acceptable.

3.3.4 Availability of SI Pumps with Pressurizer Level ≤ 5%

To mitigate a loss of decay heat removal event, TS 3.5.4.2 requires at least one SI pump and flow path to be available in either Mode 5 or Mode 6 with the pressurizer level less than or equal to 5%. The licensee evaluated this from an LTOP perspective. The licensee's evaluation concluded that should an SI pump be inadvertently started, operators would have longer than 10 minutes (licensee calculations show 18.4 minutes) before the pressurizer would become water solid. This would provide sufficient time to credit manual operator action to mitigate this event before overpressurization would occur.

The licensee further stated that typical operating practices at Byron and Braidwood ensure that only one SI pump is available. However, for circumstances where both SI pumps are available, administrative controls ensure that operators would have to take at least three independent manual actions to start more than one pump. These manual actions include moving the hand switch for each SI pump from the Pull-to-Lock position to the Run position and opening the SI pump cold leg discharge isolation valves. Therefore from an overpressure protection standpoint, the potential for an inadvertent start of both SI pumps is not considered a credible event.

The staff finds the licensee's evaluation acceptable based on (1) the time available for operators to take manual action to mitigate the event of a single SI pump start, and (2) the administrative controls for ensuring that the inadvertent start of both SI pumps event is not credible.

3.3.5 Bolt-up Temperature

The licensee did not include instrumentation uncertainties in the bolt-up temperature. However, the licensee proposed to account for instrumentation uncertainties in the minimum pressurization temperature (Section 2.5 of each unit's PTLR) to ensure that the RCS is not capable of being pressurized (i.e., the RCS remains vented) until the RCS temperature is greater than or equal to the minimum allowable bolt-up temperature plus instrument uncertainty as determined using a technique consistent with ISA S67.04 - 1994. This was determined to be acceptable by the staff.

4.0 CONCLUSIONS

Based upon the staff evaluations, as discussed in Section 3.0 above, the NRC concludes that it is acceptable for the licensee to relocate the P/T limit curves and LTOP system limits from the Byron, Units 1 and 2, and Braidwood, Units 1 and 2, TS to a licensee controlled PTLR. The staff also concludes that it is acceptable to relocate the surveillance capsule withdrawal schedule to the licensee controlled PTLR since changes to this schedule are controlled by the requirements of Appendix H to 10 CFR Part 50.

The staff has reviewed the proposed fluence values for the four Byron and Braidwood units for the revision of the P/T curves and finds that the existing approved values are conservative and, therefore, acceptable. In addition, the P/T limits meet the requirements of 10 CFR Part 50, Appendix G and none of the RT_{PTS} values exceed the screening criteria specified in 10 CFR 50.61.

The NRC notes, however, that the licensee should (1) re-evaluate the appropriate method for determining the best-estimate chemical composition as additional chemical composition data become available, (2) assess the impact of its assumption that the vessel weld has the same

$RT_{NDT(U)}$ value as determined from the surveillance weld in future revisions to the PTLR and/or when additional surveillance data become available, and (3) address the method for assessing the credibility of the data including the method for accounting for irradiation environment and chemical composition differences in future revisions to the PTLR.

The staff has also reviewed the licensee's implementation of WCAP-14040 with regard to LTOP and the licensee's proposed LTOP system enable temperature and actuation setpoints. The staff finds the licensee's proposal consistent with the staff's SE approving the WCAP and also consistent with BTP RSB 5-2, ASME Code Case N-514 and ASME Section XI, Appendix G, 1996 Addenda. Based on the above discussion and the evaluation provided in Section 3 of this SE, the staff finds the licensee's proposed LTOP enable temperature and actuation setpoints acceptable. The staff further finds acceptable the licensee's request to use the RELAP5/MOD2-B&W computer code for LTOP analyses.

5.0 REFERENCES

1. Letter from R. R. Assa, NRC, to D. L. Farrar, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60 - Braidwood Station, Unit 1," July 13, 1995.

2. Letter from C. I. Grimes, NRC, to R. A. Newton, Westinghouse Electric Corporation, "Acceptance for Referencing of Topical Report WCAP-14040, Revision 1, 'Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,'" October 16, 1995. (Also known as WCAP-14040-NP-A).

3. Letter from G. F. Dick, NRC, to I. M. Johnson, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60, 'Acceptance Criteria for Fracture Prevention Measures for Lightwater Nuclear Power Reactors for Normal Operation' - Byron Station, Units 1 and 2," November 29, 1996.

4. Letter from J. B. Hosmer, Commonwealth Edison Company, to NRC Document Control Desk, "Reactor Vessel Integrated Surveillance Program 10 CFR 50, Appendix H, Section III.C," May 6, 1997. (WCAP-14824, Revision 1 is Attached).

5. Letter from J. B. Hosmer, Commonwealth Edison Company, to NRC Document Control Desk, "Application for Amendment to Appendix A, Technical Specifications, for Facility Operating Licenses Relocation of Pressure and Temperature Limits," May 21, 1997.

6. Letter from J. B. Hosmer, Commonwealth Edison Company, to NRC Document Control Desk, "Supplement to the Application for Amendment to Appendix A, Technical Specifications, for Facility Operating Licenses Relocation of Pressure and Temperature Limits," November 18, 1997. (WCAP 14940 and Errata to WCAP-14940 and WCAP-14970 are Attached).

7. Letter from J. B. Hosmer, Commonwealth Edison Company, to NRC Document Control Desk, "Supplemental Information Pertaining to Byron & Braidwood's Reactor Vessel Integrated Surveillance Program," December 3, 1997. (WCAP-14824, Revision 2, and Erratum to WCAP-14824, Revision 2 are Attached).

8. Letter from G. F. Dick, NRC to I. M. Johnson, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60 - Braidwood Nuclear Station, Unit 2," December 12, 1997.

9. Letter from H. G. Stanley, Commonwealth Edison Company, to NRC Document Control Desk, "Supplemental Information Pertaining to Technical Specification Amendment Regarding Pressure Temperature Curves - Byron and Braidwood Nuclear Power Stations," January 8, 1998. (Errata to WCAP-14824 Revision 2, WCAP-14940 and WCAP-14970 are Attached).

10. Letter from H. G. Stanley, Commonwealth Edison Company, to NRC Document Control Desk, "Supplemental Information Pertaining to Technical Specification Amendment Regarding Pressure Temperature Curves - Byron and Braidwood Nuclear Power Stations," January 13, 1998.

11. Letter from G. F. Dick, NRC, to O. D. Kingsley, Commonwealth Edison Company, "Exemption from Requirements of 10 CFR 50.60 - Byron, Units 1 and 2, and Braidwood, Units 1 and 2," January 16, 1998.

12. Letter from R. A. Capra, NRC, to O. D. Kingsley, Commonwealth Edison Company, "Integration of Reactor Pressure Vessel Surveillance Program for Byron and Braidwood, Units 1 and 2," January 16, 1998.

Principal Contributors: K. Karwoski
L. Lois
M. Shuaibi
M. W. Weston

Date: January 20, 1998