Commonwealth Edison
1400 Opus Place
Downers Grove, Illinois 60515

March 30, 1994

Mr. William Russell, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Application for Amendment to Facility Operating Licenses-Reactor Coolant System

Braidwood Station Units 1 and 2
NPF-72/77; NRC Docket Nos. 50-456/457

Dear Mr. Russell:

Pursuant to 10 CFR50.90, Commonwealth Edison Company (CECo) proposes to amend Appendix A, Technical Specifications of Facility Operating Licenses NPF 72 and NPF 77. The proposed amendment requests changes to Technical Specifications Sections 3.4.9.1 and 3.4.9.3 and the bases Section 3/4.4.9.

The proposed amendment request consists of changes to: 1) the reactor vessel heat-up and cool-down curves, 2) the surveillance capsule removal schedule, 3) the overpressure protection system setpoint, and 4) the associated Technical Specification Bases.

The amendment request is subdivided as follows:

Attachment A: Description and Safety Analysis of Proposed Changes

Attachment B: Proposed Revision to the Technical Specifications

Attachment C: Evaluation of Significant Hazards Considerations

Attachment D: Environmental Assessment

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The proposed changes have been reviewed and approved by the On-site and Off-site Review Committees in accordance with CECo procedures. CECo has reviewed this proposed amendment in accordance with 10 CFR 50.92(c) and has determined that no significant hazards consideration exists.

CECo is notifying the State of Illinois of our application for these amendments by transmitting a copy of this letter and the associated attachments to the designated State Official.

Braidwood Unit 1 is expected to exceed 4.5 Effective Full Power Years during the month of July, 1994; therefore, CECo requests that the review and approval of the proposed amendment to be completed by June 30, 1994.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects these statements are not based on my personal knowledge, but on information furnished by other CECo employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Please address any further comments or questions regarding this matter to this office.

OFFICIAL SEAL
MARY JO YACK
NOTARY PUBLIC STATE OF ILLINOIS
MY COMMISSION EXPIRES 11/29/97

Mary Jo Yack 3/30/94

Sincerely,

Denise M. Saccomando

Denise M. Saccomando
Nuclear Licensing Administrator

Attachments

cc: R. R. Assa, Braidwood Project Manager - NRR
S. G. Dupont, Senior Resident Inspector - Braidwood
B. Clayton, Branch Chief - Region III
Office of Nuclear Facility Safety - IDNS

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ATTACHMENT A

DESCRIPTION AND SAFETY ANALYSIS OF
PROPOSED CHANGES TO APPENDIX A
TECHNICAL SPECIFICATIONS OF
FACILITY OPERATING LICENSES
NPF-72 AND NPF-77

DESCRIPTION OF THE PROPOSED CHANGE

Commonwealth Edison Company (CECo) proposes to revise Technical Specification (TS) 3.4.9.1, Reactor Coolant System, Pressure/ Temperature Operating Limits, TS 3.4.9.3, Reactor Coolant System, Overpressure Protection Systems, and TS Bases 3/4.4.9. This revision will consist of changes to reactor vessel heat-up and cool-down curves and to cold overpressure protection setpoint curves to take into account results of surveillance capsule analysis, changes to surveillance capsule removal schedule, and changes to TS Bases consistent with the aforementioned TS changes.

The revised Technical Specification pages indicating the proposed changes are provided in Attachment B.

1. Technical Specification 3.4.9.1, Heatup and Cooldown

Description and Bases of the Current Requirement

The current temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Appendix G.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the 1973 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code.

Heatup and cooldown limit curves are calculated using the most limiting value of reference nil-ductility temperature (RT_{NDT}) at the end of 32 effective full power years (EFPY)for Unit 1 (revised in Amendment 30 to 4.5 EFPY for cooldown and 12 EFPY for heatup in accordance with Regulatory Guideline 1.99, Revision 2 (RG 1.99, Rev. 2)) and 16 EFPY for Unit 2 for the reactor vessel. These applicability time frames were chosen such that the limiting RT_{NDT} at the 1/4 vessel wall thickness (1/4T) location in the core region is greater than RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the reactor coolant system will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, the most limiting RT_{NDT} based upon fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 (shown in Appendix 1) and the largest value of ΔRT_{NDT} computed by either RG 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials" or the Westinghouse Copper Trend Curves shown in Figure B 3/4.4-2 (Appendix 1). The heatup and cooldown limit curves of Figures 3.4-2a, 3.4-2b, 3.4-3a, and 3.4-3b include predicted adjustments for this shift in RT_{NDT} at the end of 4.5 EFPY and 12 EFPY (heatup and cooldown, respectively) for Unit 1 and 16 EFPY for Unit 2. The heatup and cooldown curves also include adjustments for possible errors in the pressure and temperature sensing instruments.

Values of RT_{NDT} determined in the above manner may be used until results from the material surveillance program, evaluated according to American Society for Testing and Materials (ASTM) E185, are available. Specimen removal table 4.4-5 contains lead factors representing the relationship between the fast neutron flux density at the location for the capsule and the inner wall of the reactor vessel. The results obtained from surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} from the equivalent radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR 50, and these methods are discussed in detail in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves," April 1975.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of 1/4T, and length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in the Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of RT_{NDT} is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period from which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Boiler and Pressure Vessel Code, Section III, Division 1. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp\left[0.0145(T-RT_{NDT} + 160)\right]$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\,K_{IM} + K_{IT} \leq K_{IR}$$

Where:

K_{IM} = the stress intensity factor caused by membrane (pressure) stress

K_{IT} = the stress intensity factor caused by the thermal gradients

K_{IR} = constant provided by the code as a function of temperature relative to the RT_{NDT} of the material

C = 2.0 for level A and B service limits

C = 1.5 for inservice hydrostatic and leak test operations

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for the RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{IT}, for the reference flaw are computed. The pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

For the calculation of the allowable pressure versus coolant temperature during cooldown, the reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

The use of the composite curve in the cooldown analysis is necessary because control of the cooldown procedure is based on the measurement of reactor coolant temperature, whereas the limiting pressure is actually dependent on the material temperature at the tip of the assumed flaw. During cooldown, the 1/4T vessel location is at a higher temperature than the fluid adjacent to the vessel ID. This condition, of course, is not true for the steady state situation. It follows that at any given reactor coolant temperature, the ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{IT}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and ensures conservative operation of the system for the entire cooldown period.

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the wall that alleviates the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T during steady state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the affects of compressive thermal stresses and lower K_{IR}'s do not offset each other, and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to ensure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T deep outside surface flow is assumed. Unlike the situation at the inside surface, the thermal gradients established at the outside during heatup produce stresses which are tensile in nature and therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitation because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling conditions switch from the inside to the outside, and the pressure limits must at all times be based on analysis of the most critical criterion.

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

<u>Description and Bases of the Requested Revision</u>

CECo has reanalyzed the Unit 1 heatup and cooldown based on the surveillance capsule U results. The limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated using approved methods as discussed below.

The newly calculated heatup and cooldown curves bound those currently in the Technical Specifications and comply with NRC RG 1.99, Rev. 2, for determining RT_{NDT} values at 1/4t and 3/4T locations.

Heatup and cooldown limit curves are calculated using the most limiting value of RT_{NDT} for the reactor vessel. The most limiting RT_{NDT} of the material in the core region of the reactor vessel is determined by using the pre-service reactor vessel material fracture toughness properties and estimating the radiation-induced ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60 °F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the original unirradiated RT_{NDT}. The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steel. The NRC has published a method for predicting radiation embrittlement in RG 1.99, Rev. 2. RG 1.99, Rev. 2, is used for the calculation of RT_{NDT} values at 1/4T and 3/4T locations.

The fracture-toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the NRC Regulatory Standard Review Plan (NUREG 0800, 1981). The pre-irradiation fracture-toughness properties of Braidwood Unit 1 reactor vessel are contained in Table 1, below.

Table 1
Braidwood Unit 1 Reactor Vessel Toughness Table
(Unirradiated)

COMPONENT	Cu (%)	Ni (%)	°F
Closure head flange	--	--	-20
Vessel flange	--	--	-10
Intermediate Shell Forging 49D383/ 49C344-1-1	.05	.73	-30
Lower Shell Forging 48D867/ 49C813-1-1	.03	.73	-20
Circumferential Weld WF562	.04	.67	40

The 1983 Amendment to 10 CFR 50, Appendix G, has a rule which addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the total temperature of the closure flange regions must exceed the material RT_{NDT} by at least 120 °F for normal operation when the pressure exceeds 20% of the pre-service hydrostatic test pressure.

Table 1 indicates that the initial RT_{NDT} of -10°F occurs in the vessel flange of Braidwood Unit 1, so the minimum allowable temperature of this region is 110°F. These limits are shown in revised Figures 3.4-2a and 3.4-3a (Attachment B) whenever applicable.

Limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated using the NRC approved methods. Figure 3.4-2a is the heatup curve for 100°F/hr and is applicable up to 32 EFPY with margins for possible instrumentation errors. Figure 3.4-3a is the cooldown curve up to 100°F/hr and is applicable up to 32 EFPY with margins for possible instrumentation errors. Both figures are shown in Attachment B.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 3.4-2a and 3.4-3a. This is in addition to other criteria which must be met before the reactor is made critical.

The leak test limit curve shown in Figure 3.4-2a represents minimum temperature requirements at the leak test pressure specified by NUREG-0800, LWR Edition, 1981, Standard Review Plan "Fracture Toughness Requirements" and ASME Boiler and Pressure Code, Section III, Division 1, Appendix G. The leak test curve was determined by methods of NUREG 0800 and 10 CFR 50, Appendix G. Figures 3.4-2a and 3.4-3a defined limits for ensuring prevention of nonductile failure of the Braidwood Unit 1 Primary Reactor Coolant System.

From RG 1.99, Rev. 2, the adjusted reference temperature (ART) for each material in the beltline is given by the following expression:

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin}$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 Section III of the ASME Boiler and Pressure Vessel Code. Measured values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and should be calculated as follows:

$$\Delta RT_{NDT} = [CF]f^{(0.28-0.10 \log f)}$$

To calculate ΔRT_{NDT} at any depth (e.g., at 1/4t or 3/4T), the following formula must first be used to attenuate the fluence at the specific depth.

$$f_{(depth\ x)} = f_{surface}(e^{-.24x})$$

where x (in inches) is the depth into the vessel wall measured from the vessel inner (wetted) surface. The resultant fluence is then put into the ΔRT_{NDT} equation, above, to calculate ΔRT_{NDT} at the specific depth.

CF (°F) is the chemistry factor, obtained from RG 1.99, Rev. 2. All materials in the beltline region of Unit 1 were considered for the limiting material. RT_{NDT} at 1/4T and 3/4T are summarized in Table 2. From Table 2, it can be seen that the limiting material is intermediate to the lower shell weld for heatup/cooldown curves applicable up to 32 EFPY.

Table 2
Braidwood Unit 1
Summary of Adjusted Reference Temperature; 1/4T and 3/4T Location

COMPONENT	32 EFPY RT_{NDT} at 1/4T	32 EFPY RT_{NDT} at 3/4T
Intermediates Shell	40 °F	25 °F
Lower Shell	27 °F	15 °F
Intermediate to Lower Shell Weld	159 °F *	135 °F *

* These RT_{NDT} numbers used to generate heatup and cooldown curves applicable up to 32 EFPY.

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. The limits of heatup and cooldown, specified in Technical Specifications, are required to be calculated periodically using NRC approved methods.

In addition to beltline materials, Appendix G of 10 CFR 50 also imposes pressure/temperature limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the pre-service system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the reference temperature of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. The flange reference temperature of -10°F located in Table 1, may also be found in Table B 3/4.4-1a of Braidwood's Technical Specifications.

Section IV.B of Appendix G requires that the predicted Charpy upper shelf energy (USE) at end of life be above 50 ft-lb. The unirradiated Charpy USE is 80 ft-lb for the upper to lower shell girth weld metal. This is also found in Table B 3/4.4-1a of the Technical Specifications.

Impact of the Changes

The pressure-temperature curves provide reactor coolant system (RCS) limits to protect the reactor pressure vessel from brittle fracture by clearly separating the region of normal operations from the region where the vessel is subject to brittle fracture. The heatup and cooldown limits are designed to ensure that the 10 CFR 50 Appendix G Pressure Temperature limits for the RCS are not exceeded during any condition of normal operation including anticipated operational occurrences.

The new curves do not effectively represent any appreciable change in the current methodologies; they merely provide continued assurance, up to 32 EFPY, that the Reactor Coolant System is protected from brittle fracture. The new operating limits were generated with industry standards and regulations (ASME Code Section III, and NRC RG 1.99, Rev. 2) which are recognized as being inherently conservative.

2. Proposed Changes to Surveillance Capsule Removal Schedule for Units 1 and 2

Description and Bases of the Current Requirement

A method for performing analyses to guard against fast fracture in reactor pressure vessels has been presented in "Protection Against Nonductile Failure," Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The method uses fracture mechanics concepts and is based on the RT_{NDT}.

RT_{NDT} and, in turn, . \perating limits of nuclear power plants can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement changes in mechanical properties of a given reactor pressure vessel steel can be monitored by a reactor surveillance program such as the Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program, in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens are tested. The increase in the average Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is used to index the material to the K_{IR} curve and, in turn, to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials.

Surveillance Capsules are removed in accordance with the requirements of ASTM E185-73 and 10 CFR Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5 (Attachment B). The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict the future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Description and Bases of the Requested Revision

Unit 1 and Unit 2 each have six surveillance capsules for monitoring the effects of neutron exposure on the reactor vessel. The six capsules were positioned in the reactor vessel between the neutron shield pads and the vessel wall prior to initial startup. The vertical center of the capsules is opposite the vertical center of the core.

Capsule U was removed at 1.10 EFPY for Unit 1 and 1.15 EFPY for Unit 2. The specimens contained Charpy V-notch, tensile, and 1/2T compact tension specimens from the lower shell forgings, 49D867-1/49C813-1 for Unit 1 and 50D102-1/50C97-1 for Unit 2, and weld metal representative of the intermediate to lower shell beltline weld seam of the reactor vessel and Charpy V-notch specimens from weld heat-affected zone material. All heat-affected zone (HAZ) specimens were obtained from within the HAZ of forging 49D867-1/49C813-1 (50D102-1/50C97-1 for Unit 2) of the representative weld. The proposed revision to Table 4.4-5 is a result of the capsule U removal and testing and is contained within Section 7.0 of WCAP-12685, August 1990, and WCAP-12845, March 1991, for Unit 1 and Unit 2, respectively. This updated removal schedule meets ASTM E185-82, was projected based on the design base fluence calculations and is recommended for future capsules to be removed from the Braidwood Unit 1 and Unit 2 reactor vessels.

Impact of the Changes

The removal schedules given in revised Table 4.4-5 were based on the calculated design basis neutron flux levels given in Table 6-1 of WCAP 12685 and WCAP 12845. If the units operate with low leakage fuel management strategy, the neutron flux at the vessel wall and the capsule locations would be reduced. The fluence levels reached at the target EFPYs would be correspondingly decreased.

For plants similar in design to Braidwood, operation with low leakage fuel management typically results in neutron fluence rates at the capsule locations of approximately $2.7 \times 10^{18} n/cm^2$-EFPY and $3.0 \times 10^{18} n/cm^2$-EFPY at the 29° and 31.5° capsule locations, respectively. These values can be used to readjust withdrawal schedules to accommodate low leakage operation.

Implementing the revised schedule will change the lead factors and withdrawal times of the capsules. Also, for Unit 2, Capsule X, instead of Capsule W, will be withdrawn during the next refueling outage and Capsule W will be scheduled as a standby capsule. The proposed removal schedule meets ASTM E185-82, is based on the design basis fluence calculations, and is recommended by Westinghouse, via WCAP 12685 and WCAP 12845, for future capsule removal from Braidwood Units 1 and 2.

3. Technical Specification 3.4.9.3, Overpressure Protection Systems

Description and Bases of the Current Requirement

The setpoints provided for the Low Temperature Overpressure Protection System (LTOPS) are selected such that the pressure peaks resulting from design basis overpressure events are limited to values less than those specified by Appendix G of 10 CFR 50. Appendix G provides the fracture toughness requirements for reactor vessels under specified operating conditions, and RG 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials" specifies the procedure acceptable to the NRC staff for calculating the pressure limits required by Appendix G.

Description and Bases of the Requested Revision

The analysis of capsule U, obtained from the Braidwood Unit 1 reactor vessel, resulted in a revision to the current pressure-temperature limits. As a result of these new limits, Westinghouse was instructed by Commonwealth Edison Company to develop new LTOPS setpoints. The revised LTOPS setpoints are presented in the new Figure 3.4-4a contained in Attachment B.

Westinghouse also recommended an LTOPS enable (during cooldown) or disable (during heatup) temperature of 310 °F. The recommendation is based on the intersection of the pressurizer power operated relief valves (PORVs) normal operational setpoint of 2335 psig and the pressure-temperature limit specified by Appendix G. This provides pressure relief capability throughout the full range of temperatures spanned by heatup and cooldown operation. For Braidwood Unit 1 at 32 EFPY, the intersection (hence, the enable temperature recommendation) is at approximately 310 °F. Since the pressure-temperature limits have been calculated for a reactor vessel exposure of 32 EFPY, the enable temperature recommendation is conservative throughout the life of the plant, assuming the continued applicability of the current algorithm used to generate the pressure-temperature limits.

The standard initial condition assumed for a low temperature overpressure analysis is that the reactor coolant system is water solid, and that the transient results from either a mass injection or a heat input event. The mass injection transient is based on the operation of a single centrifugal charging pump, and is initiated by a spurious loss of letdown concurrent with a failure of the charging flow controls to the maximum flow allowed by the reactor coolant system pressure. The heat injection event results from the start of a reactor coolant pump assuming that the shell side of the steam generator is 50 °F warmer than the water contained in the reactor coolant system. This represents the maximum temperature asymmetry that could develop during a cooldown maneuver, following the shutdown of the reactor coolant pump, and the continued cooling by means of the residual heat removal system. Isothermal conditions are assumed to exist between the primary and secondary sides of the steam generators, so that when an RCP is started, the steam generator's secondary side acts as a heat source for the NSSS. The setpoint selection is based on the most restrictive of either the mass injection or the heat input cases, without credit for the residual heat removal system's safety valves.

Impact of the Changes

The setpoint program shown graphically in Figure 3.4-4a of Attachment B, is based on a steady-state pressure-temperature limit corresponding to a reactor vessel exposure of 32 Effective Full Power Years (EFPY). In addition, the setpoint function accounts for a 50 °F thermal transport effect, a 27 °F temperature streaming and instrumentation uncertainty, and the 800 psig PORV piping limit. Consistent with current Westinghouse practice, nominal pressure values are assumed; i.e., pressure instrument uncertainty is not included in the analysis.

The setpoints have been selected to prevent opening both pressurizer PORVs at the same time, while providing reactor coolant pump No. 1 seal protection. Simultaneous opening of both PORVs would result in a large underpressure that would place the RCP No. 1 seal at risk. As a result of this analysis, PORV setpoints could not be selected above the residual heat removal system (RHR) safety valve setpressure value of 495 psig (450 psig plus 10% accumulation).

The administratively controlled practice of Braidwood Unit 1, via 1BwGP 100-5, "Plant Shutdown and Cooldown," is to arm low temperature overpressure protection (LTOPS) prior to decreasing below 350 °F. This practice will continue to be conservative with respect to the Westinghouse recommended setpoint of 310 °F.

4. Technical Specification Bases 3/4.4.9 Pressure/Temperature Limits

Description and Bases of the Current Requirement

The current bases for pressure/temperature limits contains a footnote and discussion, added in Amendment 30, of the applicability of the current heatup and cooldown curves. The applicability of the Braidwood Unit 1 curves was revised from 32 EFPY to 4.5 EFPY in accordance with RG 1.99, Rev. 2.

Description and Bases of the Requested Revision

This proposed change deletes references to Braidwood Unit 1 heatup and cooldown limitations based on 4.5 EFPY which will be superseded upon approval of this amendment request.

Impact of the Changes

This proposed change, to delete references to 4.5 EFPY restrictions on Braidwood Unit 1 reactor vessel is editorial in nature. The change specifies that Unit 1 heatup and cooldown curves are applicable to 32 EFPY.

E. SCHEDULE REQUIREMENTS

The proposed amendment to Technical Specifications 3.4.9.1 and 3.4.9.3 are required for Braidwood Unit 1 to exceed 4.5 EFPY. Braidwood Unit 1 is scheduled to reach 4.5 EFPY during the month of July 1994. Therefore, Braidwood requests that this amendment be approved by June 30, 1994. This amendment submittal has been deferred by the Licensee pending resolution of a potential issue with Westinghouse which may affect this amendment. This issue, differential pressure between Reactor Vessel beltline region and the pressure sensing instruments located in the Hot Leg of the Reactor Coolant System when the Reactor Coolant Pumps are running, has not been completely resolved with the vendor. Thus, administrative controls will continue to govern compliance with Appendix G limits.