ATTACHMENT A

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 127

The following is a list of the affected pages:

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B 3/4 4-15a
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MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

INITIAL $RT_{NDT}$ : 60°F

$RT_{NDT}$ AFTER 10 EFPY: 1/4T, 140°F 128

3/4T, 129°F

15

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY. ~~CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.~~

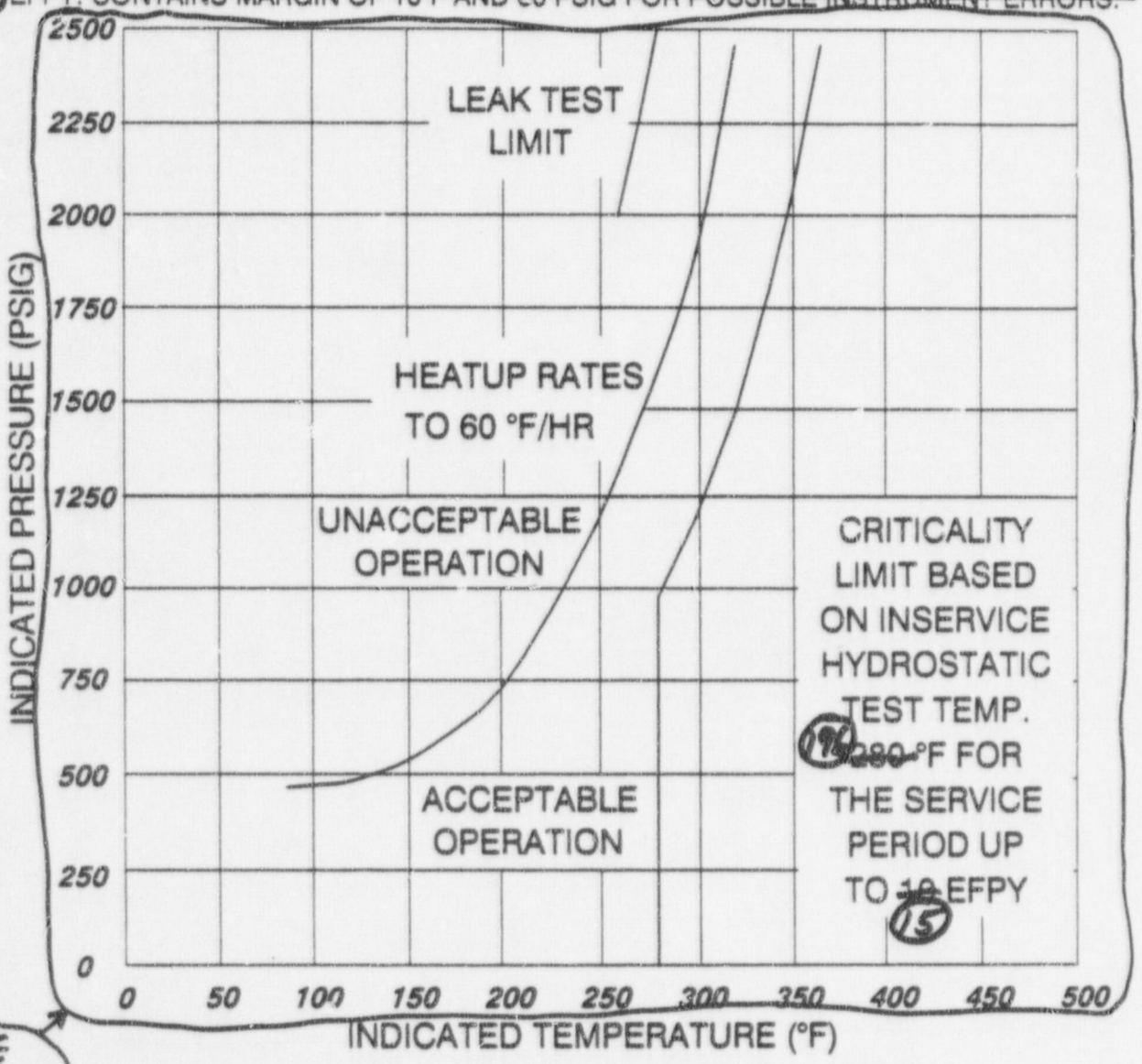


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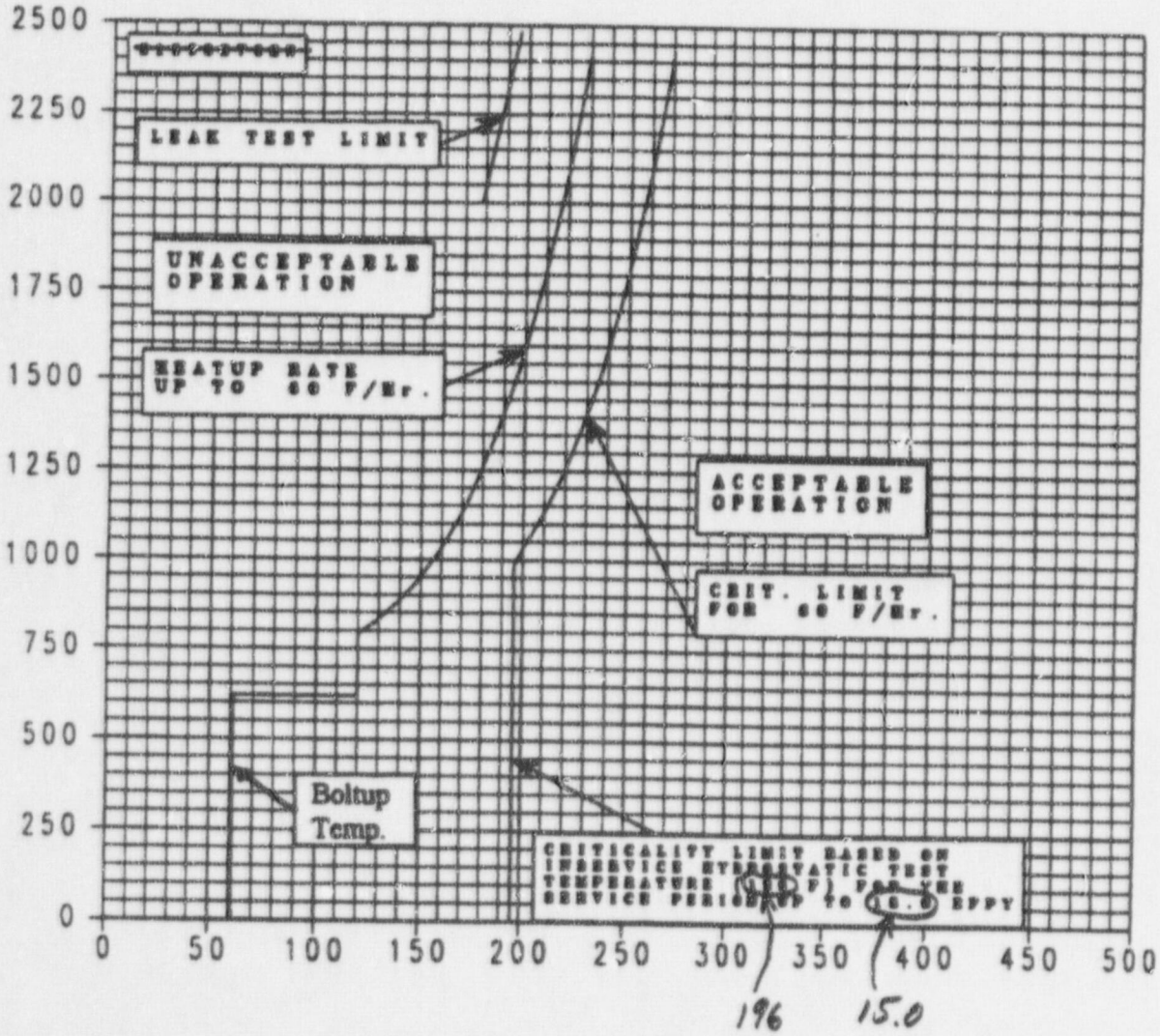
**FIGURE 3.4-2**

**Beaver Valley Unit 2 Reactor Coolant System Heatup Limitations Applicable for the First 10 EFPY** 15



(Proposed Wording)

INSERT A


2500
2250
2000
1750
1500
1250
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400
450
500
LEAK TEST LIMIT
UNACCEPTABLE OPERATION
HEATUP RATE UP TO 60 F/Hr.
ACCEPTABLE OPERATION
CRIT. LIMIT FOR 60 F/Hr.
Boltup Temp.
CRITICALITY LIMIT BASED ON INSERVICE HYDROSTATIC TEST TEMPERATURE (196 F), FOR THE SERVICE PERIOD UP TO (15.0 EFPY
196
15.0

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: INTERMEDIATE SHELL PLATE B9004-1

INITIAL $RT_{NDT}$: 60°F

$RT_{NDT}$ AFTER 10 EFPY: 1/4T, 140 °F (128)
3/4T, 129 °F

15

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 10 EFPY. ~~CONTAINS MARGIN OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS.~~

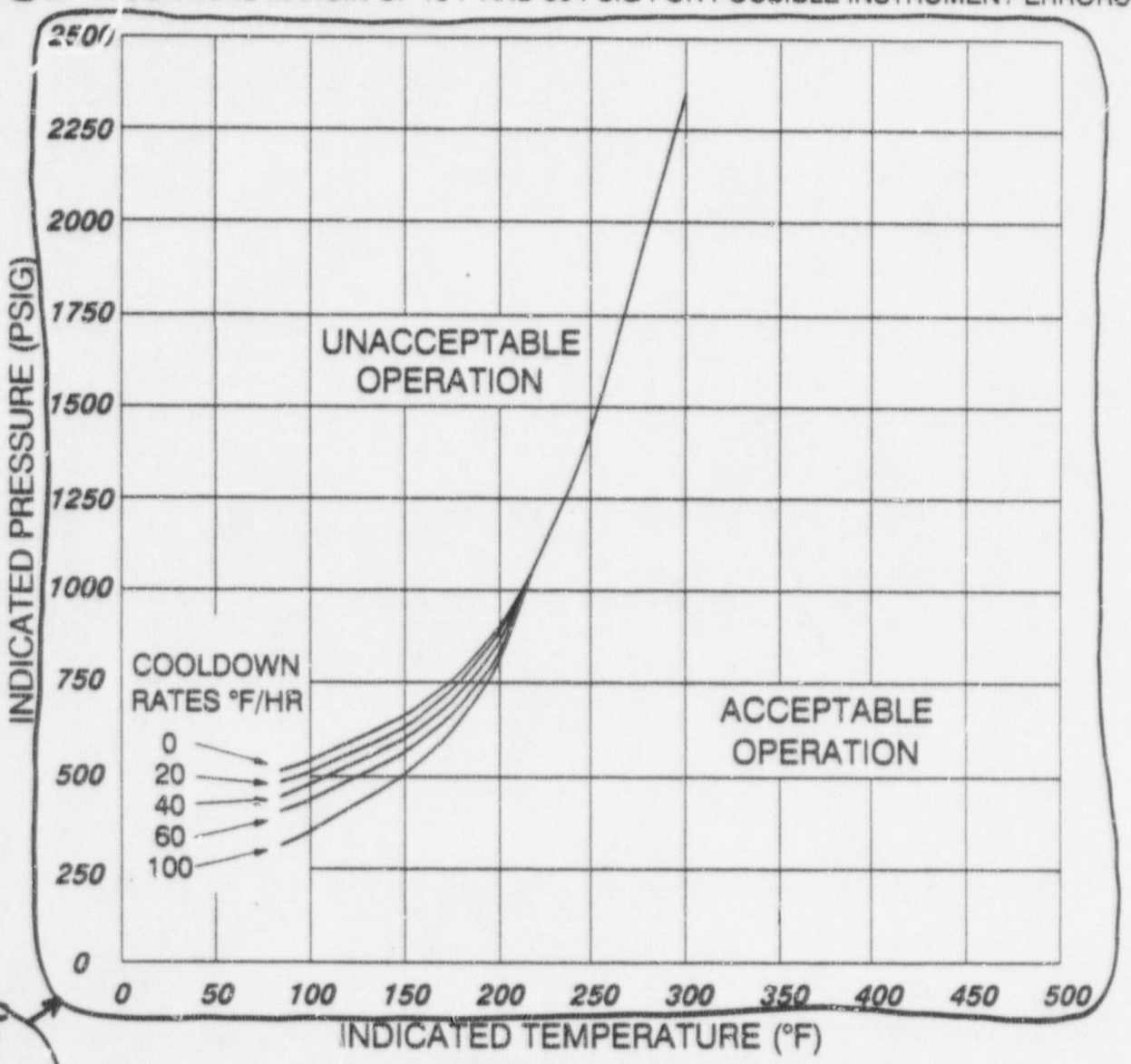


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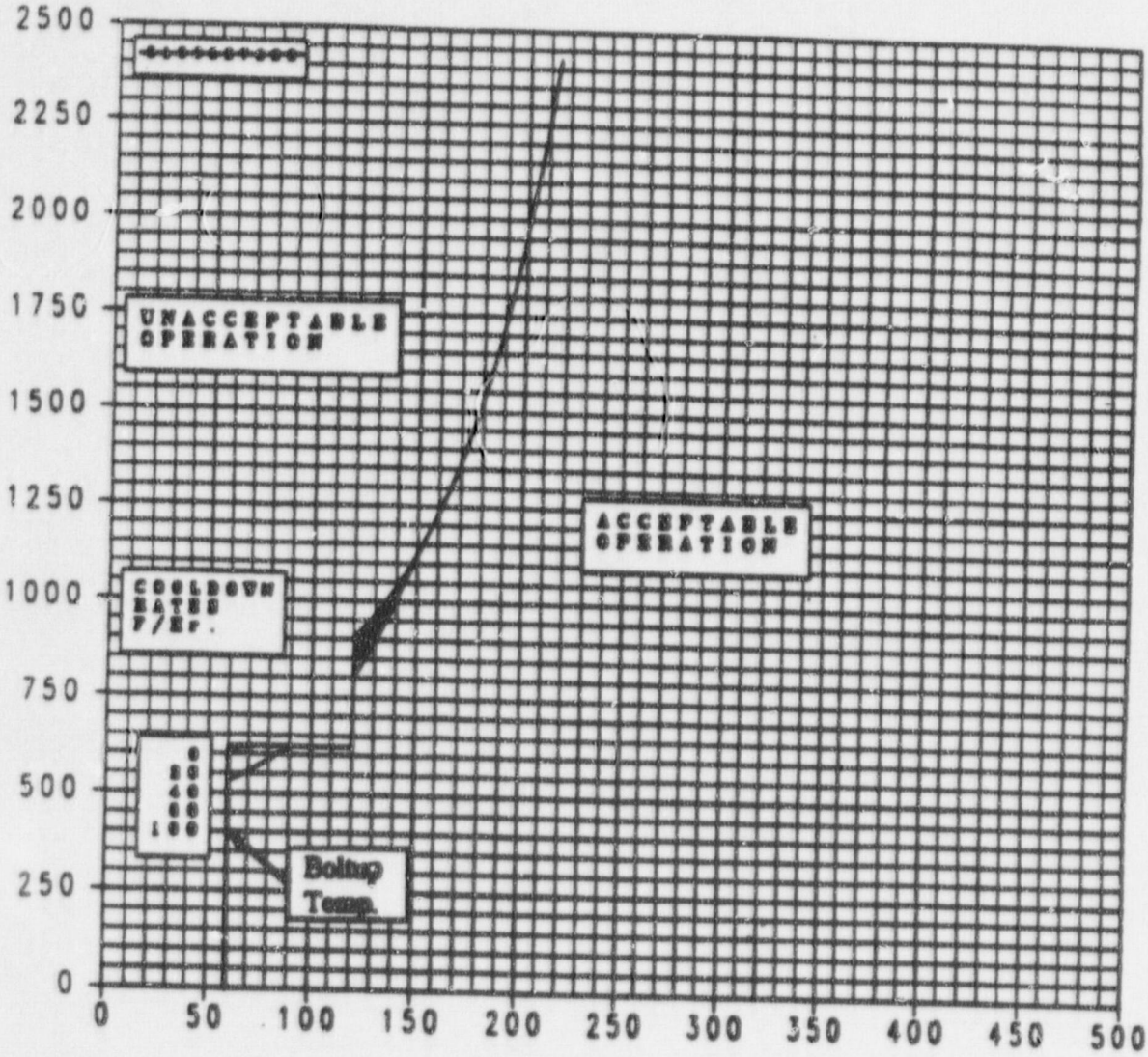
**FIGURE 3.4-3**
**Beaver Valley Unit 2 Reactor Coolant System Cooldown Limitations Applicable for the First 10 EFPY** 15


(Proposed Wording)

INSERT B




Separate each cooldown rate into a separate figure, i.e., (a, b, c, d and e)

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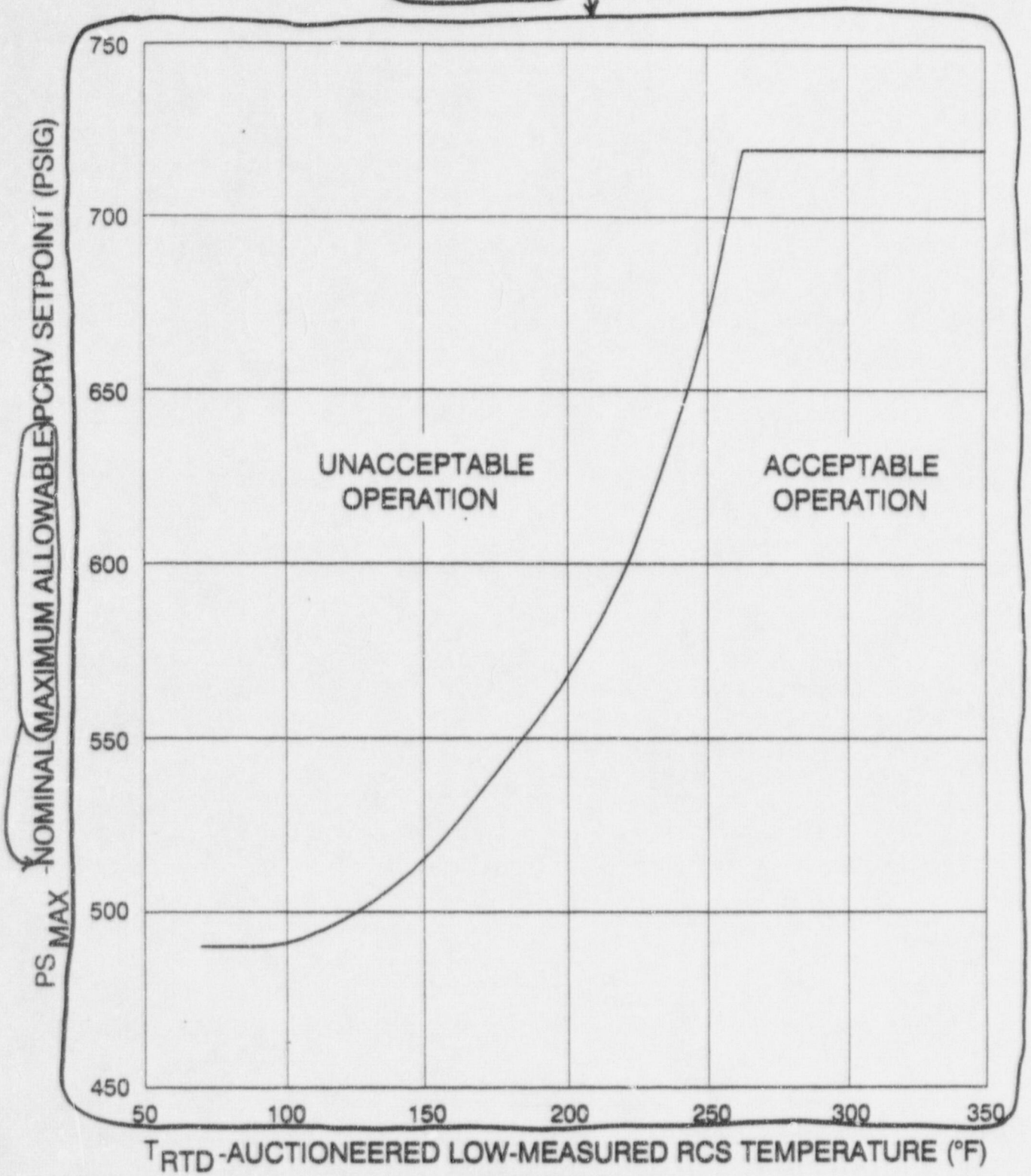


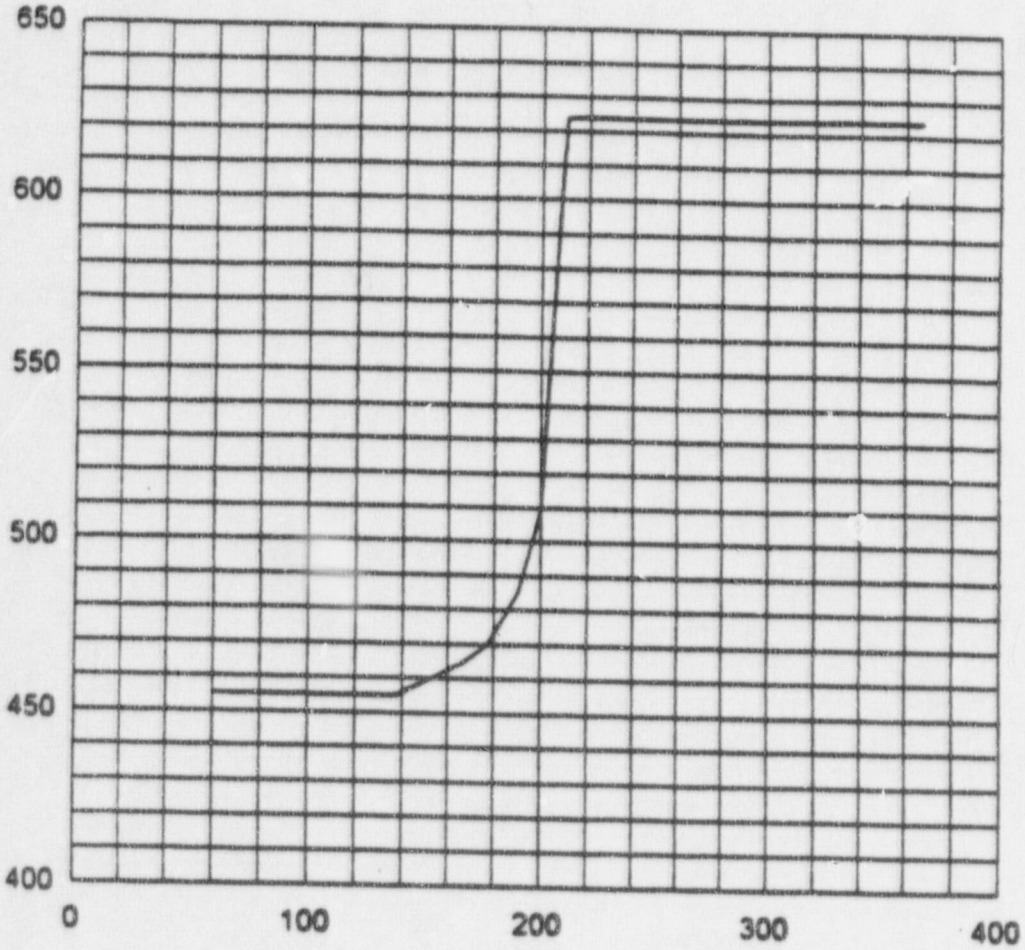

**FIGURE 3.4-4**

**MAXIMUM ALLOWABLE NOMINAL PORV SETPOINT FOR THE OVERPRESSURE PROTECTION SYSTEM**



(Proposed wording)

INSERT C



(a, b, c, d and e)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 60°F per hour. The cooldown limit curves, Figure 3.4-3, are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 10 EFPY. 15

nickel WCAP-15179

The reactor vessel materials have been tested to determine their initial $RT_{NDT}$; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E >1 Mev) irradiation will cause an increase in the $RT_{NDT}$. Therefore, an adjusted reference temperature, based upon the fluence, copper content and phosphorus content of the material in question, can be predicted using Figures B 3/4.4-1 and Regulatory Guide 1.99, Revision 1, "~~Effects of Residual Elements on Predicted~~ Radiation Damages to Reactor Vessel Materials." The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3, include predicted adjustments for this shift in $RT_{NDT}$. ~~as well as adjustments for possible errors in the pressure and temperature sensing instruments. Additionally, these curves are not impacted by the special 10 CFR Part 50 rules for closure flange regions due to the low initial $RT_{NDT}$ of the flange material.~~ Embrittlement of

Heatup and cooldown limit curves are calculated using the most limiting value of $RT_{NDT}$ (reference nilductility temperature). The most limiting $RT_{NDT}$ of the material in the core region of the reactor vessel is determined by using the preservice reactor vessel material properties and estimating the radiation-induced $\Delta RT_{NDT}$. $RT_{NDT}$ is designated as the higher of either the drop weight nilductility transition temperature ($T_{NDT}$) 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.

nickel

$RT_{NDT}$ increases as the material is exposed to fast neutron radiation. Thus, to find the most limiting $RT_{NDT}$ at any time period in the reactor's life, $\Delta RT_{NDT}$ 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 phosphorus) present in reactor vessel steels. The Regulatory Guide 1.99 Revision 1 curves which show the effect of fluence, copper content and phosphorus content on $\Delta RT_{NDT}$ for reactor vessel steels are shown in Figure B 3/4.4-2.

and 1 2 on upper shelf energy (USE)



(Proposed Wording)

TABLE B 3/4.4-1
REACTOR VESSEL TOUGHNESS DATA (1)

| COMPONENT | CODE NO. | MATERIAL SPEC. NO. | Cu % | Ni % | P % | $T_{NDT}$ °F | 50 FT/LB 35 MIL TEMP °F | $RT_{NDT}$ °F | USE FT-LBS. |
|---|---|---|---|---|---|---|---|---|---|
| Closure Head Dome | B9008-1 | A533B, CL.1 | .13 | .54 | .013 | -20 | 50 | -10 | 137 |
| Closure Head Flange | B9002-1 | A508, CL.2 | --- | .74 | .012 | -10 | <40 | -10 | 136 |
| Vessel Flange | B9001-1 | A508, CL.2 | --- | .22 | .010 | 0 | <10 | 0 | 132.5 |
| Inlet Nozzle | B9011-1 | A508, CL.2 | --- | .88 | .006 | 0 | <10 | 0 | 104 |
| Inlet Nozzle | B9011-2 | A508, CL.2 | --- | .88 | .010 | 10 | <10 | 10 | 115 |
| Inlet Nozzle | B9011-3 | A508, CL.2 | --- | .84 | .009 | 20 | <40 | 20 | 122 |
| Outlet Nozzle | B9012-1 | A508, CL.2 | --- | .71 | .007 | -10 | < 0 | -10 | 137 |
| Outlet Nozzle | B9012-2 | A508, CL.2 | --- | .74 | .006 | -10 | < 0 | -10 | 121 |
| Outlet Nozzle | B9012-3 | A508, CL.2 | --- | .68 | .008 | -10 | < 0 | -10 | 112 |
| Nozzle Shell | B9003-1 | A533B, CL.1 | .13 | .61 | .008 | -10 | 110 | 50 | 91 |
| Nozzle Shell | B9003-2 | A533B, CL.1 | .12 | .58 | .009 | 0 | 120 | 60 | 79.5 |
| Nozzle Shell | B9003-3 | A533B, CL.1 | .13 | .61 | .008 | -10 | 110 | 50 | 97.5 |
| Inter. Shell | B9004-1 | A533B, CL.1 | .07 | .57 | .010 | 0 | 120 | 60 | 83 |
| Inter. Shell | B9004-2 | A533B. CL.1 | .07 | .59 | .007 | -10 | 100 | 40 | 75.5 |
| Lower Shell | B9005-1 | A533B, CL.1 | .08 | .59 | .009 | -50 | 88 | 28 | 82 |
| Lower Shell | B9005-2 | A533B, CL.1 | .07 | .58 | .009 | -40 | 93 | 33 | 77.5 |
| Bottom Head Torus | B9010-1 | A533B, CL.1 | .15 | .49 | .007 | -30 | 56 | -4 | 97 |
| Bottom Head Dome | B9009-1 | A533B, CL.1 | .14 | .50 | .007 | -30 | 35 | -25 | 116 |
| Weld (Inter. & Lower Shell Long. Seams & Girth Seam)* | | | .08 | .07 | .008 | -30 | <30 | -30 | 144.5 |
| HAZ (Plate B9004-2) | | | --- | | ---- | -80 | 40 | -20 | 76 |

*Same heat of wire and lot of flux used in all seams including surveillance weldment.

1. For evaluation of Inservice Reactor vessel Irradiation damage assessments, the best estimate chemistry values reported in the latest responce to Generic Letter 92-01 or equivalent document are applicable.



(Next page is B 3/4 4-10)
(Proposed wording)

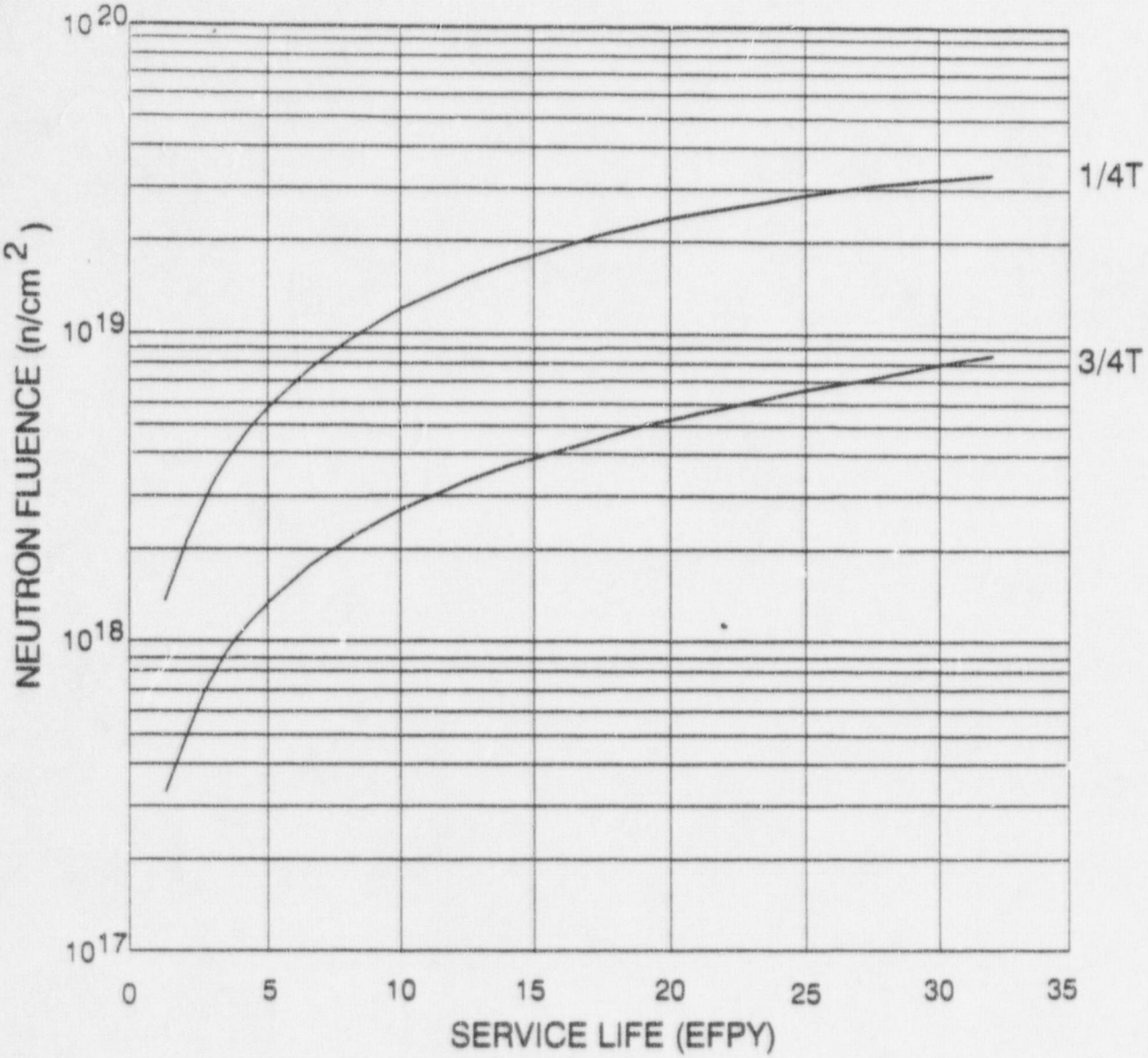



**FIGURE B 3/4 4-1**

**FAST NEUTRON FLUENCE (E>1 Mev) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)**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REPLACE WITH INSERT D



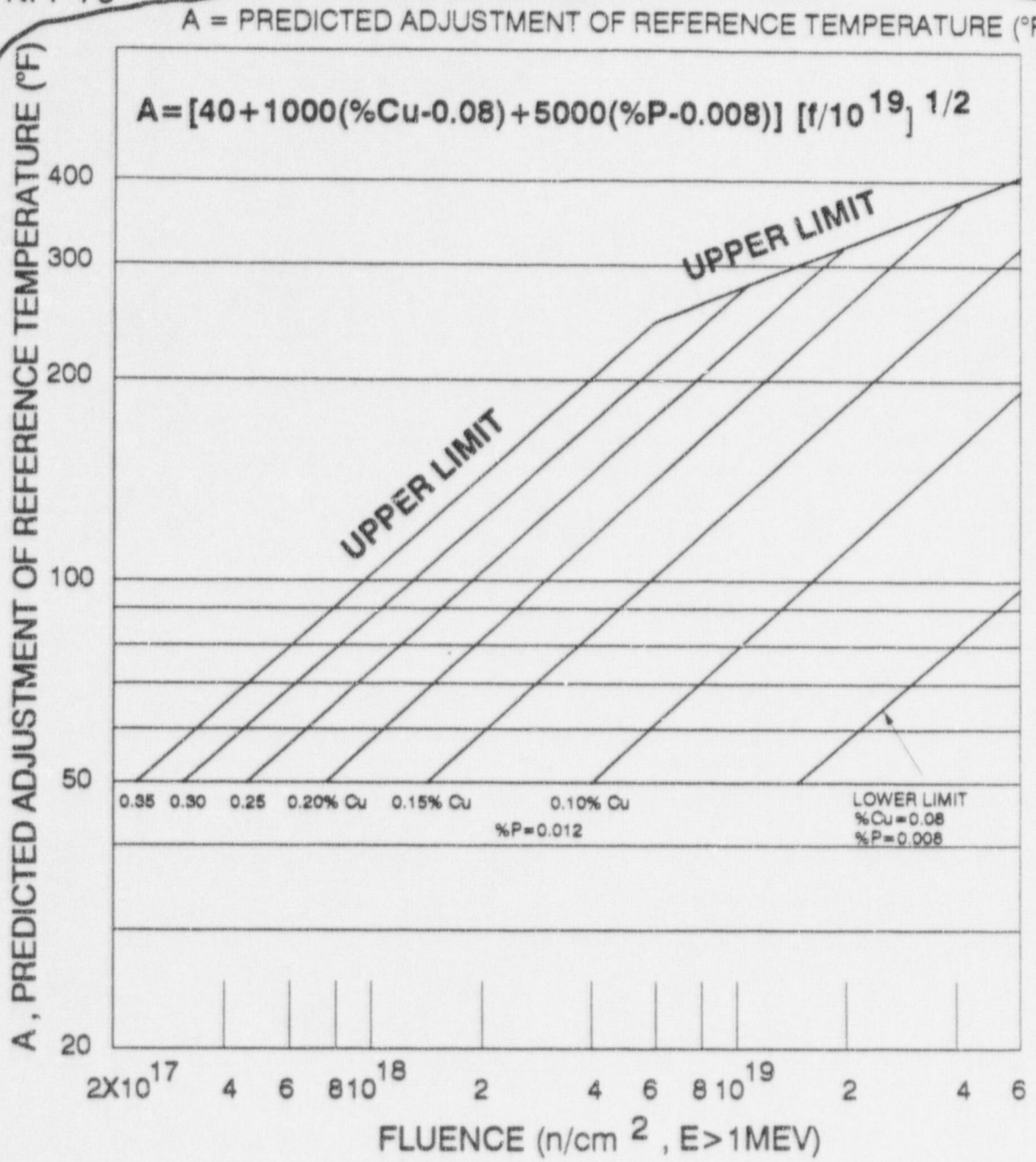


**FIGURE B 3/4 4-2**

**PREDICTED ADJUSTMENT OF REFERENCE TEMPERATURE, "A", AS A FUNCTION OF FLUENCE AND COPPER CONTENT FOR COPPER AND PHOSPHORUS CONTENT OTHER THAN THOSE PLOTTED, USE THE EXPRESSION FOR "A" GIVEN ON THE FIGURE**



(Proposed wording) ~~REISSUED MARCH 92~~

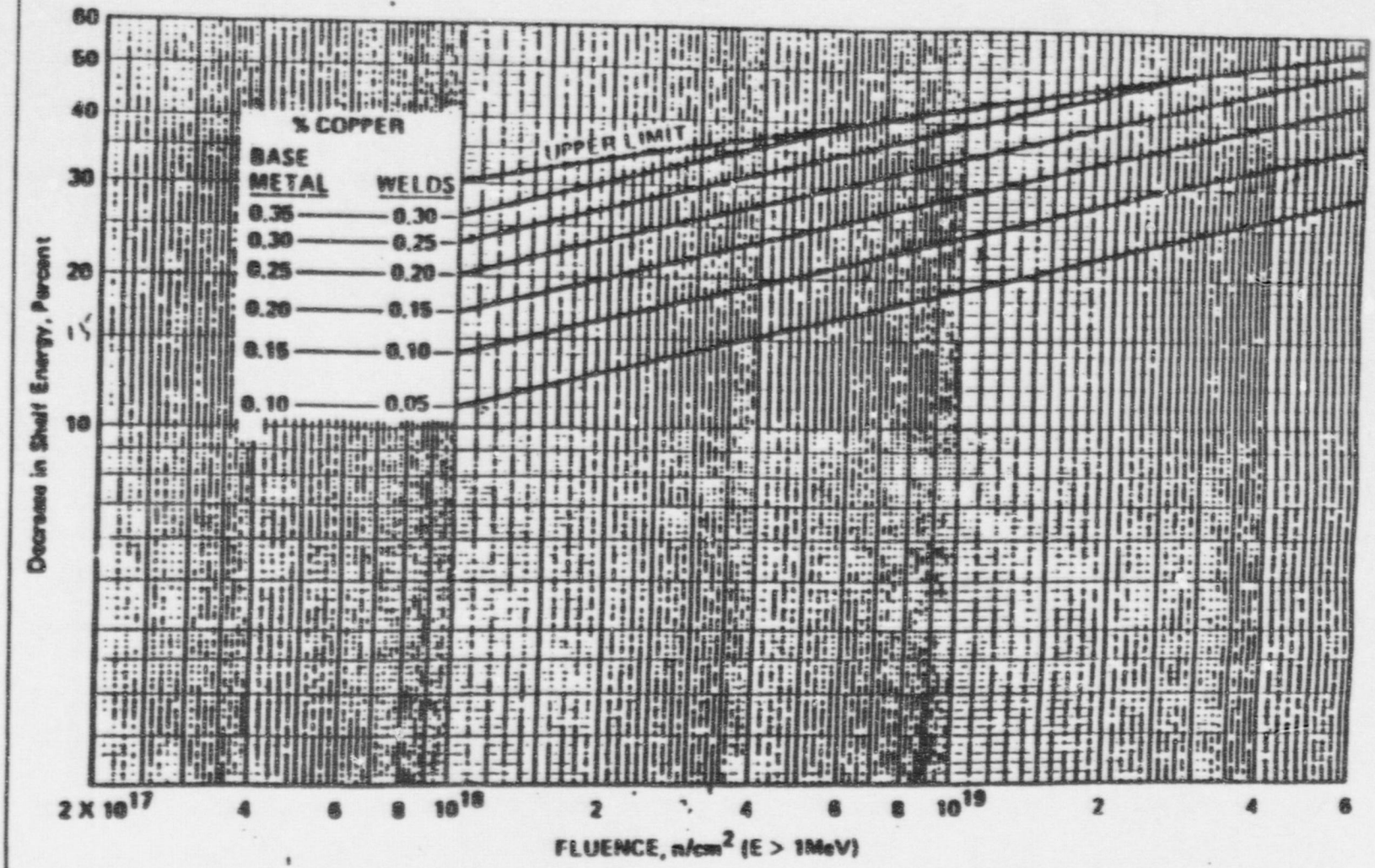


FIGURE 2 Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence

Figure B 3/4 4-1

INSERT D

REACTOR COOLANT SYSTEM

predicted by the equation: $\Delta RT_{NDT} = (CF) f (0.28 - 0.1 \log f)$

can be generated

BASES

nickel

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

Given the copper and phosphorous contents of the most limiting material, the radiation-induced $\Delta RT_{NDT}$ can be estimated from Figure B 3/4.4-2. Fast-neutron fluence (E > 1 MeV) at the 1/4 T (wall thickness) and 3/4 T (wall thickness) vessel locations are given as a function of full-power service life. ~~in Figure B 3/4.4-1.~~ The data for all other ferritic materials in the reactor coolant pressure boundary are examined to insure that no other component will be limiting with respect to $RT_{NDT}$.

The preirradiation fracture-toughness properties of the Beaver Valley Unit 2 reactor vessel materials are presented in Table B 3/4.4-1. The fracture toughness properties of the ferritic material in the reactor coolant pressure boundary are determined in accordance with the 1972 Summer Addenda to Section III of the ASME Boiler and ~~Vessel~~ Pressure Code.

INSERT E

$K_{IC}$

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 and cooldown cannot be greater than the reference stress intensity factor, $K_{IR}$, for the metal temperature at that time. $K_{IR}$ is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code.[2] The $K_{IR}$ curve is given by the equation:

by Code Case N-640

$$K_{IR} = 26.78 + 1.223 \exp [0.0145 (T - RT_{NDT} + 160)] \quad (4\text{-}1)$$

where $K_{IR}$ is the reference stress intensity factor as a function of the metal temperature T and the metal reference nilductility temperature, $RT_{NDT}$. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G to the ASME Code[2] as follows:

$$K_{IC} = 33.2 + 20.734 * e [0.02 (T - RT_{NDT})]$$

$$C\, K_{IM} + K_{It} \leq K_{IR} \quad (4\text{-}2)$$

where

$K_{IC}$

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

$K_{It}$ is the stress intensity factor caused by the thermal gradients

$K_{IR}$ is a function of temperature to the $RT_{NDT}$ of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

relative

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~~[2] ASME Boiler and Pressure Vessel Code, Section III, Division 1 - Appendices, "Rules for Construction of Nuclear Vessels," Appendix G. "Protection Against Nonductile Failure," pp. 559-569, 1980 Edition, American Society of Mechanical Engineers, New York, 1983.~~



(Proposed Wording)

Attachment A
Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 127

INSERT E

The pressure-temperature limit curves are developed using Code Case N-640. One of the safety margins incorporated into the curves is the lower bound fracture toughness curve. The lower bound fracture toughness curves available in Appendix G to Section XI use the reference stress intensity factor $K_{IA}$. The pressure-temperature limit curves based on Code Case N-640 use the reference stress intensity factor $K_{IC}$. $K_{IA}$ is a fracture toughness curve which is a lower bound on all static, dynamic and arrest fracture toughness, and $K_{IC}$ is a fracture toughness curve which is a lower bound on static fracture toughness only. The only change that is made when generating the revised pressure-temperature limits curve with $K_{IC}$ is the lower bound fracture toughness curve selected. All other margins involved in the generation process remain unchanged. Since the heatup and cooldown process is a very slow one, with the fastest rate allowed being 100°F per hour, the rate of change of pressure and temperature is considered constant so that the stress is essentially constant. Both heatup and cooldown correspond to static loading, with regard to fracture toughness. The only time when dynamic loading can occur and where the dynamic/arrest toughness $K_{IA}$ should be used for the reactor pressure vessel is when a crack is running. This might happen during a pressurized thermal shock event, but not during heatup and cooldown. Therefore, the static toughness $K_{IC}$ lower bound toughness is used to generate the pressure-temperature limit curves.

at the 1/4 T and 3/4 T location

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

At any time during the heatup or cooldown transient, $K_{IR}$ ($K_{IC}$) is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for $RT_{NDT}$, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity factors, $K_{It}$, for the reference flaw are computed. From equation 4-2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

Cooldown

For the calculation of the allowable pressure-versus-coolant temperature during cooldown, the Code reference flaw 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 increases 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 ~~ensures the~~ 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 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 1/4 T 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 $K_{IR}$ ($K_{IC}$) at the 1/4 T 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 1/4 T location and, 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 insures conservative operation of the system for the entire cooldown period.

HEATUP

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 1/4 T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the $K_{IR}$ ($K_{IC}$) for the 1/4 T crack during heatup is


(Proposed wording)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

lower than the $K_{IR}$ for the 1/4 T crack during steady-state conditions at the same coolant temperature. $K_{IC}$

$K_{IC}$'s

During heatup, especially at the end of the transient, conditions may exist such that the effects 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 1/4 T flaw is considered. Therefore, both cases have to be analyzed in order to insure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

temperature

The second portion of the heatup analysis concerns the calculation of pressure-~~temeprature~~ limitations for the case in which a 1/4 T deep outside surface flaw is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus 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 as follows: A composite curve is constructed 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 the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion. ~~Then, composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.~~

The actual shift in $RT_{NDT}$ of the vessel material will be established periodically during operation by removing and evaluating, in accordance with 10 CFR 50 Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the $\Delta RT_{NDT}$ determined from the surveillance capsule is different from the calculated $\Delta RT_{NDT}$ for the equivalent capsule radiation exposure.

, as documented in WCAP-15139,



(Proposed Wording)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing. ← (INSERT F)

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 5.3-6 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on the pressurizer heatup and cooldown rates and auxiliary spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

Pressure-temperature limit curves shown in figure B 3/4 4-~~3~~ 2 were developed for the limiting ferritic steel component within an isolated reactor coolant loop. The limiting component is the steam generator channel head to tubesheet region. This figure provides the ASME III, Appendix G limiting curve which is used to define operational bounds, such that when operating with an isolated loop the analyzed pressure-temperature limits are known. The temperature range provided bounds the expected operating range for an isolated loop and Code Case N-640

OVERPRESSURE PROTECTION SYSTEMS

BACKGROUND

The overpressure protection system (OPPS) controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G. The reactor vessel is the limiting RCPB component for demonstrating such protection. The maximum setpoint for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup meet the 10 CFR 50, Appendix G requirements during the OPPS MODES.

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

INSERT F

These pressure-temperature limit lines on Figures 3.4-2 and 3.4-3 (a, b, c, d and e) for boltup temperature are provided to ensure compliance with the minimum temperature requirements of Appendix G to ASME Section XI for vessel closure head flange boltup. It recommends that when the flange and adjacent shell region are stressed by the full intended bolt preload the minimum metal temperature in the stressed region is at least the initial $RT_{NDT}$ temperature for the material in the stressed regions.

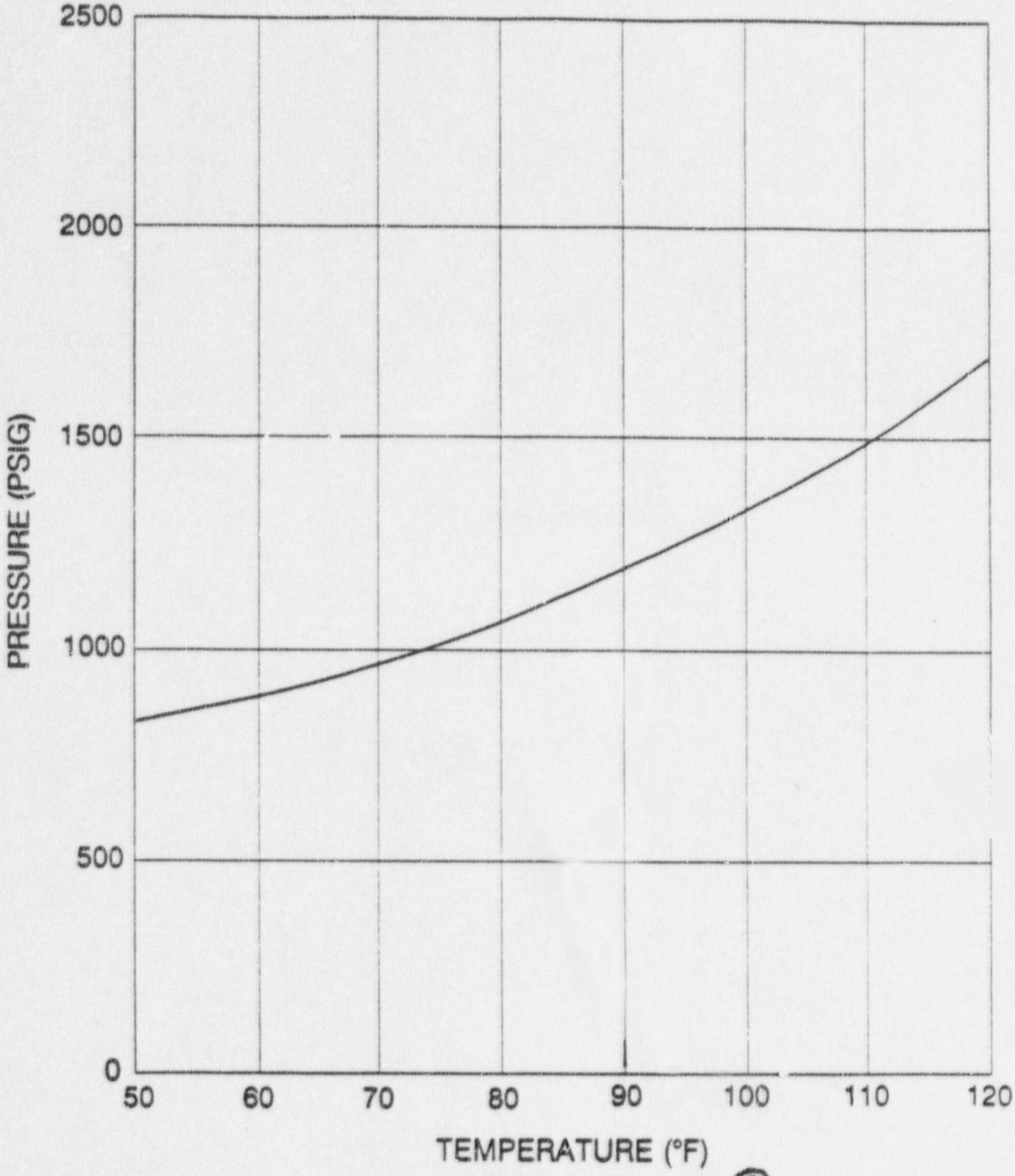


**FIGURE B 3/4 4-3** 2

**ISOLATED LOOP PRESSURE-TEMPERATURE LIMIT CURVE**



(Proposed Wording)

~~Added by NRC letter dated 9/2/92~~

BASES (Continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

BACKGROUND (Continued)

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 brittle cracking 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 limits.~~

3.4.9.3, "Overpressure Protection Systems"

~~This~~ LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires deactivating all but one charging pump and isolating the 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 safety injection (SI) actuation circuits blocked. Due to the lower pressures in the OPPS MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve and, if needed, until the charging pump is actuated by SI.

The OPPS for pressure relief consists of two PORVs with reduced lift settings or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurization for the required coolant input capability.

PORV REQUIREMENTS

As designed for the OPPS System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the OPPS actuation logic. The OPPS actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable in the limits is approached. 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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

PORV REQUIREMENTS (Continued)

a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open. Having the setpoints of both valves within the limits ensures that the Appendix G limits will not be exceeded in any analyzed event. When a PORV is opened in an increasing pressure transient, the release of coolant 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.

RCS VENT REQUIREMENTS ← INSERT G

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 OPPS 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 may be satisfied by removing a pressurizer safety valve or establishing an opening between the RCS and the containment atmosphere of the required size through any positive means available which cannot be inadvertently defeated. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses demonstrate that the reactor vessel is adequately protected against exceeding the P/T limits when low RCS temperature conditions exist. At the enable temperature and below, overpressure prevention is provided by two OPERABLE RCS relief valves or a depressurized RCS and a sufficient sized RCS vent.

The actual temperature at which the pressure in the P/T limit curve falls below the OPPS setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the heatup and cooldown curves are revised, the OPPS must be re-evaluated to ensure its functional requirements can still be met.

INSERT G

The low limit on pressure during the transient is typically established based solely on an operational consideration for the Reactor Coolant Pump (RCP) No. 1 seal to maintain a nominal differential pressure across the seal faces for proper film-riding performance. The upper limit (based on the minimum of the steady-state 10 CFR 50 Appendix G requirement and the PORV piping limitations) and the RCP No. 1 seal performance criteria create a pressure range from which the setpoints for both PORVs are selected. When there is insufficient range between the upper and lower pressure limits to select the PORV setpoints to provide protection against violating both limits, setpoint selection to provide protection against the upper limit violation takes precedence.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEAT INPUT TYPE TRANSIENTS (Continued)

The analyses demonstrate that either one RCS relief valve or the depressurized RCS and RCS vent can maintain the RCS pressure below the limits when only one charging pump is actuated by SI. Thus, the LCO allows only one charging pump OPERABLE during the OPPS MODES. Since neither one RCS relief valve nor the RCS vent can handle a full SI actuation, the LCO also requires the accumulators isolated.

The isolated accumulators must have their discharge valves closed with power removed. Fracture mechanics analyses established the temperature of OPPS Applicability at the enable temperature.

PORV PERFORMANCE

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit. The setpoint is derived by analyses that model the performance of the OPPS assuming the limiting transient of SI actuation of one charging pump. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the P/T limits will be met.

Nominal allowable nominal

The Maximum Allowed PORV Setpoint for the OPPS is derived by analysis which models the performance of the OPPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G limits will not be violated with consideration for: (1) a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening; (2) a 50°F heat transport effect made possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for OPPS; (3) instrument uncertainties; ~~and~~ (4) single failure.

reactor vessel

The PORV setpoint will be updated when the revised P/T limits conflict with the OPPS analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.9.1, "Pressure/Temperature Limits," discuss these examinations.

; and (5) the pressure difference between the wide range pressure transmitter and the reactor vessel limiting beltline region

/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

PORV PERFORMANCE (Continued)

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RCS VENT PERFORMANCE

With the RCS depressurized, analyses show that a PORV or equivalent opening with a vent size of 3.14 square inches is capable of mitigating the allowed OPPS overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the OPPS configuration, SI actuation with one charging pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

The RCS vent size is based on the PORV size, therefore, the vent is bounded by the PORV analyses.

The RCS vent is passive and is not subject to active failure.

LCO

This LCO requires that the OPPS is OPERABLE. The OPPS is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the limits as a result of an operational transient.

INSERT H

To limit the coolant input capability, the LCO requires one charging pump capable of injecting into the RCS and all accumulator discharge isolation valves closed and immobilized. The LCO is qualified by a note that permits two pumps capable of RCS injection for less than or equal to 15 minutes to allow for pump swaps. This note also allows all charging pumps capable of injecting into the RCS during a change from MODE 3 to MODE 4 to be OPERABLE for a limited period of time.

The LCO is also qualified by a note stating that accumulator isolation with power removed from the discharge isolation valves is only required when the accumulator pressure is greater than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This note permits the accumulator discharge isolation valve surveillance to be performed only under these pressure and temperature conditions.

INSERT H

The Maximum Allowable Nominal Setpoint Curve defines the maximum nominal setpoint at which the PORVs can be set which will ensure that Appendix G limits are not exceeded. To maximize operating margin, the setpoint for the higher PORV is set at the Maximum Allowable Nominal Operating Curve within the respective instrumentation loop calibration tolerance band. The PORV setpoint uncertainty is calculated with reference to the methodology in ISA 67.04-1994 for performing instrumentation uncertainty calculations. The instrumentation calibration tolerances are provided in plant procedures. The overall setpoint calculation accounts for the instrumentation calibration tolerances in the uncertainty calculation.

Since actuation of both PORVs can result in excessive undershoot below the PORV setpoint, the lower PORV setpoints are staggered by an amount greater than or equal to the limiting overshoot (from either the mass injection or heat addition events). The staggered setpoints are provided in plant procedures.

ATTACHMENT B

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 127
UPDATE HEATUP AND COOLDOWN CURVES AND OVERPRESSURE PROTECTION SYSTEM (OPPS) SETPOINT CURVE

## A. DESCRIPTION OF AMENDMENT REQUEST

The proposed amendment would extend the applicability of the heatup and cooldown curves of Technical Specification (TS), Section 3.4.9.1, Pressure/Temperature (P/T) Limits. A new Figure 3.4-2 provides the Beaver Valley Unit 2 Reactor Coolant System Heatup Limitation Applicable for the First 15 Effective Full Power Years (EFPY). A new Figure 3.4-3 (a, b, c, d and e) provides the Beaver Valley Unit 2 Reactor Coolant System Cooldown Limitation Applicable for the First 15 EFPY. The proposed changes include new heatup and cooldown curves developed in accordance with the methodology provided in Regulatory Guide (RG) 1.99 Revision 2 (Reference 1) and Code Case N-640 using the Surveillance Capsule V results. The applicability of Section 3.4.9.3, Overpressure Protection Systems, is also updated to 15 EFPY. A new Figure 3.4-4 provides the Maximum Allowable Nominal PORV (power operated relief valves) Setpoint for the Overpressure Protection System.

The methodology for the P/T limits has been changed from RG 1.99 Revision 1 to RG 1.99 Revision 2. Bases Section B 3/4.4.9, Reactor Coolant System, has also been revised to incorporate new references and associated methodology changes. Figure B 3/4.4-1, titled "Fast Neutron Fluence as a Function of Full Power Service Life" has been deleted since it is not used by RG 1.99 Revision 2 which replaces the previous reference to Regulatory Guide 1.99 Revision 1. Figure B 3/4.4-2, Predicted Adjustment of Reference Temperature, "A", as a Function of Fluence and Copper Content, is replaced by a new figure, Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence, since RG 1.99 Revision 2 refers to the new figure to show the effect of fluence and copper content on upper shelf life for reactor vessel steels. This figure has also been renumbered to B 3/4 4-1 and, as a result, Figure B 3/4 4-3 has been renumbered to B 3/4 4-2, and reference to these figures in the text has been modified as well. An equation has been added to page B 3/4 4-11 which is used to predict the radiation-induced shift in the reactor vessel reference temperature ($\Delta RT_{NDT}$). Previously Figure B 3/4.4-2 was used.

A new column was added to Table B 3/4.4-1, Reactor Vessel Toughness Data, for nickel since it is one of the predominant metals used to generate the Adjusted Reference Temperature (ART). A footnote was added to this table to indicate that, for evaluation of inservice reactor vessel irradiation damage assessments, the best estimate chemistry values reported in the latest response to Generic Letter (GL) 92-01 or equivalent document are applicable. Reference to $K_{IR}$ has been replaced with

$K_{IC}$ in accordance with the change to the methodology used in Code Case N-640. A description of the term "Boltup Temperature" has been added to the Bases in Insert F. The purpose of the boltup temperature limit line is to ensure compliance with the minimum temperature requirements of Appendix G to ASME Section XI for vessel flange and closure head flange boltup. Selection criteria has been added to the Bases to describe the basis used for selecting the PORV limits in Insert G. It notes that the Appendix G requirement takes precedence over the reactor coolant pump (RCP) No. 1 seal limit. A discussion of the Maximum Allowable Nominal PORV setpoint was also added to the Bases in Insert H to indicate that instrumentation calibration tolerances are included within the setpoint.

B. DESIGN BASES

The proposed changes replace the P/T limits and the OPPS setpoint curves with those applicable to 15 EFPY based on the analysis of Surveillance Capsule V. The revised heatup and cooldown curves do not account for instrument uncertainties since the OPPS setpoint curve is more conservative than the heatup and cooldown curves and is used for plant operation. As long as the OPPS curve limits are satisfied the heatup and cooldown limits are not exceeded. Instrument uncertainties have been included in the determination of the OPPS setpoints. Additionally, the OPPS setpoints conservatively account for the pressure difference between the wide-range pressure transmitter and the reactor vessel limiting beltline region. The OPPS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the reactor coolant system (RCS). According to WCAP 14040-NP-A, Rev. 2 (Reference 2), it is appropriate to utilize the steady-state Appendix G limit. WCAP-14040-NP Revision 1 was approved by the NRC by letter dated October 16, 1995, which was incorporated in Revision 2 of the WCAP issued in January 1996. In addition, OPPS also provides for an operational consideration to maintain the integrity of the PORV piping. Since the heatup and cooldown curves are generated using the $K_{IC}$ fracture toughness methodology, Code Case N-514 (which permits a 10% relaxation of the Appendix G pressure temperature limits up to the OPPS enable temperature) is not applicable. The Code Case N-640 methodology is used. The "enable" temperature, the RCS temperature below which OPPS is required to operate, is calculated in accordance with Code Case N-640.

The ability of the reactor vessel to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to significant fast neutron bombardment. Generally, the overall effects of fast neutron irradiation on the mechanical properties of low alloy, ferritic pressure vessel steels such as SA533 Grade B Class 1, which is the base metal of the reactor pressure

vessel, show an increase in hardness and tensile properties and a decrease in ductility and toughness during high energy irradiation. The Reactor Vessel Radiation Surveillance Program, designed by Westinghouse, is described in WCAP-9615, Revision 1, "Duquesne Light Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program" (Reference 3). The surveillance program was planned to cover the 40-year design life of the reactor pressure vessel and was based on ASTM E185-73, "Standard Recommended Practice for Surveillance Tests for Nuclear Vessels."

A method for performing analyses to guard against fast fracture in reactor pressure vessels has been presented in Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code (Reference 4). The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature ($RT_{NDT}$).

$RT_{NDT}$ is defined as the greater of either the drop weight nil-ductility transition temperature (NDTT) per ASTM E-208 or the temperature 60°F less than the 50 ft-lb (and 35-mil lateral expansion) temperature as determined from Charpy specimens oriented normal (transverse) to the major working direction of the plate. The $RT_{NDT}$ of a given material is used to index that material to a reference stress intensity factor curve ($K_{IC}$ curve) provided in Code Case N-640. The $K_{IC}$ curve is a lower bound static fracture toughness curve based on results obtained from several heats of pressure vessel steel. When a given material curve is indexed to the $K_{IC}$ curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors. The use of $K_{IC}$ adds substantial pressure margin to the heatup and cooldown curves than that provided by the stress intensity factor $K_{IA}$ of Reference 5. $K_{IA}$ was used for the current heatup and cooldown curves.

$RT_{NDT}$ and, in turn, the operating 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 Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program, in which a surveillance capsule is periodically removed from the reactor vessel capsule location and the encapsulated specimens tested. The increase in the average Charpy V-notch 30 ft-lb temperature, $\Delta RT_{NDT}$, due to irradiation is added to the initial $RT_{NDT}$, along with margin to cover uncertainties, to adjust the $RT_{NDT}$ for radiation embrittlement. This ART (ART = initial $RT_{NDT}$ + margin for uncertainties + $\Delta RT_{NDT}$) is used to index the material to the $K_{IC}$ curve and, in turn, to set operating limits which take into account the effects of irradiation on the reactor vessel materials.

Overpressure protection for the RCS is achieved by means of self-actuated safety valves located high in the system on the steam space of the pressurizer. These safety valves have a set pressure based on the RCS design pressure of 2485 psig and are intended to protect the system against transients initiated in the plant when the RCS is operating near its normal temperature. To avoid brittle fractures at reactor vessel metal temperatures below the OPPS enable temperature, the allowable system pressure is substantially less than the normal system design pressure of 2485 psig. Therefore, overpressure mitigation provisions for the reactor vessel must be available when the RCS and, hence, the reactor vessel, are at temperatures below the OPPS enable temperature.

Normally when the RCS is at a temperature below the OPPS enable temperature, the RCS is open to the Residual Heat Removal System (RHRS) for the purposes of removing residual heat from the core, providing a path for letdown to the purification subsystem and to control the RCS pressure when the plant is operating in a water solid mode. The RHRS is provided with self-actuated water relief valves to prevent overpressure in this relatively low design pressure system (600 psig) caused either within the system itself or from transients transmitted from the RCS. The RHRS relief valves will mitigate pressure transients originated in the RCS to maximum pressure values determined by the relief valves set pressure of 450 psig plus a pressure accumulation above the set pressure dependent on the liquid volume magnitude of the transient.

The low design pressure RHRS is normally isolated from the high design pressure RCS, during reactor power operation at temperatures above 350°F, by two isolation valves in series. Therefore, the RHRS can be inadvertently isolated from the RCS by these same isolation valves. The OPPS is intended to provide overpressure mitigation for the RCS by addressing those transients which may occur when the RHRS isolation valves inadvertently close thus isolating the RHRS water relief valves from the RCS. The OPPS controls the RCS pressure at low temperature so the integrity of the Reactor Coolant Pressure Boundary (RCPB) is not compromised by violating the P/T Limits.

This evaluation will demonstrate that plant operation with the revised heatup and cooldown and OPPS curves will not adversely affect the pressure boundary integrity or operability of the RCS, and does not represent a significant hazards consideration as defined in 10 CFR 50.92.

C. JUSTIFICATION

The current heatup and cooldown curves are being revised using the results of the evaluation of Surveillance Capsule V by changing the applicability from 10 EFPY to 15 EFPY. The curves have been developed in accordance with the methodology provided

in RG 1.99 Revision 2 and Code Case N-640. The use of the Code Case N-640 $K_{IC}$ methodology, will result in an increase in the safety of the operating plant, as the likelihood of RCP seal failure and/or fuel problems will decrease. By changing from $K_{IA}$ to $K_{IC}$ methodology, the operating window relative to RCP seal requirements is wider and the probability of damaging the RCP seals and initiating a small break LOCA (a potential pressurized thermal shock initiator) are reduced. The OPPS setpoints are being revised for the same applicability. The current setpoints were based on a 20°F cooldown curve and a 60°F heatup curve. The proposed setpoints are based on isothermal conditions. The OPPS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. According to WCAP 14040-NP-A, Rev. 2, it is appropriate to utilize the steady-state Appendix G limit. In addition, the OPPS also takes into consideration those limitations required to maintain the integrity of the PORV piping.

D. SAFETY ANALYSIS

Westinghouse performed an analysis to determine curves acceptable for heatup and cooldown based on the analysis results of Surveillance Capsule V. The OPPS setpoints were then generated applicable to the 15 EFPY heatup and cooldown curves.

Heatup and cooldown curves are calculated using the most limiting value of $RT_{NDT}$ corresponding to the limiting beltline region material of the reactor vessel. The adjusted $RT_{NDT}$ of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced $\Delta RT_{NDT}$ and adding margin. The $RT_{NDT}$ is designated as the higher of either the drop weight NDTT 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 reactor life, $\Delta RT_{NDT}$ due to the radiation exposure associated with that time period must be added to the unirradiated $RT_{NDT}$. The extent of the shift in $RT_{NDT}$ is enhanced by certain chemical elements (such as copper and nickel) present in the reactor vessel steel. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in RG 1.99 Revision 2. RG 1.99, Revision 2, is used for the calculation of ART values at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region measured from the clad/base metal interface. The most limiting ART values are used in the generation of the heatup and cooldown curves.

Heatup and cooldown curves for normal heatup and cooldown have been calculated using the methods outlined in Reference 6. These limit curves are developed using Code Case N-640 as noted in Reference 7. One of the safety margins incorporated into the curves is the lower bound fracture toughness curve. The lower bound fracture toughness curves available in Appendix G to Section XI uses the reference stress intensity factor $K_{IA}$ and is the basis for the current heatup and cooldown curves. The proposed heatup and cooldown curves are based on Code Case N-640 and use the reference stress intensity factor $K_{IC}$. $K_{IA}$ is a fracture toughness curve which is a lower bound on all static, dynamic and arrest fracture toughness, and $K_{IC}$ is a fracture toughness curve which is a lower bound on static fracture toughness only. The only change in generating the revised heatup and cooldown curves with $K_{IC}$ is the lower bound fracture toughness curve selected. All other margins involved in the generation process remain unchanged. Since the heatup and cooldown process is very slow, with the fastest rate allowed being 100°F per hour, the rate of change of pressure and temperature is often constant so that the stress is essentially constant. Both heatup and cooldown correspond to static loading, with regard to fracture toughness. The only time when dynamic loading can occur and where the dynamic/arrest toughness $K_{IA}$ should be used for the reactor pressure vessel is when a crack is running. This may occur during a pressurized thermal shock event, but not during heatup and cooldown. Therefore, the static toughness $K_{IC}$ lower bound toughness is used to generate the heatup and cooldown curves. The primary reason for incorporating this change in methodology is to reduce the excess conservatism in the current Appendix G approach that could, in fact, reduce overall plant safety. By changing from $K_{IA}$ to $K_{IC}$ methodology, the operating window relative to the RCP seal requirements is wider and the chances of damaging the RCP seals and initiating a small break LOCA, a potential pressurized thermal shock initiator, are reduced.

The pressure difference between the wide-range pressure transmitter and the limiting beltline region has not been accounted for in the heatup and cooldown curves generated for normal operation. The pressure difference, however, has been incorporated into the generation of the OPPS curve and, since the OPPS setpoint curve is more conservative than the heatup and cooldown curves, and is used for plant operation, as long as the OPPS curve limits are satisfied, the heatup and cooldown limits are not exceeded.

Inserts (with a letter designation; i.e., A, B, etc.) to be added to the marked-up pages are provided in Attachment A. Insert (A) presents the heatup curve with no margin for instrumentation error using heatup rates up to 60°F/hr applicable for the first 15 EFPY. Insert (B) presents the cooldown curve with no margin for instrumentation error up to 100°F/hr applicable for the first 15 EFPY. Allowable combinations of temperature and pressure for

specific temperature change rates are below and to the right of the limit lines shown in the new curves. This is in addition to other criteria which must be satisfied prior to reactor criticality, as discussed in the following paragraphs.

Acceptable P/T limitations for reactor criticality are provided to the right of the criticality limit line shown on the new Figure 3.4-2. The straight-line portion of the criticality limit is at the minimum permissible temperature of the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Appendix G to Section XI of the ASME Code.

The criticality limit curve specifies P/T limits for core operation to provide additional margin during actual power production as specified in Reference 8. The P/T limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum permissible temperature in the corresponding P/T curve for heatup and cooldown calculated as described in Reference 6. For the heatup curve identified in Insert (A), the minimum temperature for the inservice hydrostatic leak tests for the reactor vessel at 15 EFPY is 196°F. The vertical line drawn from this point on the P/T curve, intersecting a curve 40°F higher than the P/T limit curve, constitutes the limit for core operation for the reactor vessel. The P/T limit curves define the above limits for ensuring prevention of heatup and cooldown nonductile failure for the reactor vessel.

Technical Specification 3.4.9.3 requires that OPPS be operable with a maximum of one charging pump capable of injecting into the RCS and the accumulator isolated and either:

a) Two power-operated relief valves (PORVs) with nominal maximum lift settings which vary with the RCS temperature and which do not exceed the limits established in Insert (C), or

b) The RCS depressurized and an RCS vent of greater than or equal to 3.14 square inches.

The requirements are applicable to Mode 4 when any RCS cold leg temperature is less than or equal to an enable temperature of 350°F, Mode 5, and Mode 6 when the reactor vessel head is on. With respect to the charging pumps, a footnote is provided. The footnote states that two pumps are allowed to be capable of RCS injection for less than or equal to 15 minutes for pump swaps. This note also allows all charging pumps capable of injecting into the RCS during a change from Mode 3 to Mode 4 to be operable for a limited period of time. This footnote existed prior to the proposed change to OPPS and is not affected by this change. With respect to the accumulator, a footnote is provided to state that accumulator isolation with power removed from the discharge

isolation valves is only required when the accumulator pressure is greater than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curve. This footnote is provided to ensure that the accumulator discharge isolation valve surveillance is only performed under the P/T conditions allowed by the P/T limit curve.

The OPPS curve was derived from the steady state heatup and cooldown curves at 15 EFPY. The setpoints are applicable to 30% steam generator tube plugging. The setpoints conservatively account for instrument uncertainties associated with the wide range pressure (99 psi) and wide range temperature (17°F) instrumentation, per Reference 9. The setpoints conservatively account for the pressure difference, between the wide-range pressure transmitter and the reactor vessel limiting beltline region, identified in Nuclear Safety Advisory letter NSAL-93-005A. The pressure differences according to Reference 9 are 0, 17, 30, and 47 psi for 0, 1, 2, and 3 RCPs running, respectively. The minimum value of either the adjusted Appendix G limit, after subtracting the delta-P, or the 800 psig PORV piping limit, forms the limit used as the basis for setpoint selection. Heat transport effects, which are applied to the heat injection transient results and account for a 50°F difference between the wide range temperature sensor and the reactor vessel, have also been incorporated. Pressure overshoots during the design basis events are based on a pressurizer PORV stroke open/close time of 1.65/1.0 seconds. The heatup and cooldown curves are generated using the $K_{IC}$ fracture toughness methodology; therefore, ASME Code Case N-514 is not applicable and Code Case N-640 is used. Setpoints are selected so that RCS pressures will not exceed the 15 EFPY Appendix G pressure limits down to a reactor vessel temperature of 60°F.

The design basis transient for determination of the OPPS setpoints are both the mass injection event caused by the failure of the controls for a single charging pump to go to the full flow condition and the heat addition scenario in which an RCP in a single loop is started when the RCS temperature is as much as 50°F lower than the steam generator secondary side temperature. The heat addition transient results in a sudden secondary to primary heat transfer and rapid increase in primary system pressure. The heat addition event has been analyzed for RCS temperatures between 70°F and 300°F. The influx of fluid into the relatively inelastic RCS during the mass injection event also causes a rapid increase in system pressure. The RCS temperature assumed for the mass injection event is 70°F. For any particular design basis transient, the PORV will be signaled to open at a specific pressure setpoint for a given temperature. However, there will be a pressure overshoot during the delay time before the valve starts to move and during the time the valve is moving to the full open position. This overshoot is dependent on the dynamics of the system and the input parameters and results in a maximum system pressure somewhat higher than the set pressure.

Similarly there will be a pressure undershoot, while the valve is relieving, both due to the reset pressure being below the setpoint and due to the delay in stroking the valve closed. In order to preserve the single failure criteria, the overshoots are calculated assuming the availability of one PORV during the design basis mass injection and heat addition events, when the RCS is water solid, concurrent with a loss of letdown and isolation of the RHRS. The second PORV is assumed to have failed. The maximum and minimum pressures reached in the transient are a function of the selected setpoint and fall within an acceptable pressure range.

The upper pressure limit for OPPS is defined by Appendix G requirements or on PORV piping limitations, after consideration of uncertainties and the delta-P between the wide range pressure transmitter and the reactor vessel limiting region. The lower limit on pressure during the design basis OPPS mass injection and heat addition transients is established based on operational consideration for the RCP number one seal limit which requires a nominal differential pressure across the seal faces for proper film-riding performance. As part of the OPPS setpoint evaluation, margin to the RCP number one seal limit is evaluated.

Based on the analyses of record, this limit corresponds to a differential pressure across the seal of 200 psid, which corresponds to the gage pressures provided in Reference 9. As demonstrated in Reference 9, pressure undershoot below the PORV setpoint during a design basis mass injection or heat addition event can exceed 100 psi. Therefore, with the PORV setpoints developed for the 15 EFPY heatup and cooldown curves, there is the potential for RCS pressure to violate the RCP number one seal limit at the lowest RCS temperatures.

While analysis has not been performed that models the simultaneous relief from two PORVs, undershoot below the PORV setpoint can be significantly higher if both PORVs actuate during an OPPS event, and it is anticipated that the pump seal limit would be exceeded. However, staggering the setpoints minimizes the likelihood that both PORVs will actuate simultaneously during credible OPPS events. Similarly, WCAP 14040-NP-A (Reference 2) indicates that when there is insufficient range between the upper and lower pressure limits to select PORV setpoints that provide protection against violating both limits, then the setpoint selection that provides protection against the upper limit violation takes precedence.

The final Maximum Allowable Nominal Setpoints are defined by a curve which passes through the most limiting points determined for the mass injection and heat addition events, after consideration of applicable instrumentation uncertainties and heat transport effects. Breakpoints are selected as input to the function generator so that setpoints calculated by the function generator protect the steady state Appendix G limits. These

breakpoints represent the higher PORV setpoints. Since actuation of both PORVs can result in excessive undershoot below the PORV setpoint, the lower PORV setpoints are staggered by an amount greater than or equal to the limiting overshoot (from either the mass injection or heat addition events) at setpoint temperature. In order to prevent PORV actuation should the OPPS be inadvertently armed beyond a temperature of 367°F, the setpoints ramp linearly to a maximum pressure of 2335 psig at 425°F.

E. NO SIGNIFICANT HAZARDS EVALUATION

The no significant hazard considerations involved with the proposed amendment have been evaluated. The evaluation focusing on the three standards set forth in 10 CFR 50.92(c) are as quoted below:

> The Commission may make a final determination, pursuant to the procedures in paragraph 50.91, that a proposed amendment to an operating license for a facility licensed under paragraph 50.21(b) or paragraph 50.22 or for a testing facility involves no significant hazards consideration, if operation of the facility in accordance with the proposed amendment would not:
>
> (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or
>
> (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
>
> (3) Involve a significant reduction in a margin of safety.

The following evaluation is provided for the no significant hazards consideration standards.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed heatup and cooldown curves have been revised by changing the applicability from 10 effective full power years (EFPY) to 15 EFPY. The curves have been developed in accordance with the methodology provided in Regulatory Guide 1.99 Revision 2 and Code Case N-640. The proposed heatup and cooldown curves define limits that still ensure the prevention of nonductile failure for the reactor vessel. The design basis events that were protected against have not changed; therefore, the probability of an accident is not increased.

   The overpressure protection system (OPPS) has been revised such that the applicability has changed from 10 EFPY to 15 EFPY. This system protects the Reactor Coolant System

(RCS) at low temperatures so that the integrity of the Reactor Coolant Pressure Boundary (RCPB) is not compromised by violating the pressure/temperature (P/T) limits. These changes were determined in accordance with the methodologies set forth in the regulations to provide an adequate margin of safety to ensure the reactor vessel will withstand the effects of normal cyclic loads due to temperature and pressure changes as well as the loads associated with postulated faulted events. The lower limit on pressure during the design basis OPPS mass injection and heat addition transients is established based on operational consideration for the RCP number one seal limit which requires a nominal differential pressure across the seal faces for proper film-riding performance. As part of the OPPS setpoint evaluation, margin to the RCP number one seal limit is evaluated.

This limit corresponds to a differential pressure across the seal of 200 psid, which corresponds to the gage pressures. The pressure undershoot below the PORV setpoint during a design basis mass injection or heat addition event can exceed 100 psi. Therefore, with the PORV setpoints developed for the 15 EFPY heatup and cooldown curves, there is the potential for RCS pressure to violate the RCP number one seal limit at the lowest RCS temperatures.

Undershoot below the PORV setpoint can be significantly higher if both PORVs actuate during an OPPS event, and it is anticipated that the pump seal limit would be exceeded. However, staggering the setpoints minimizes the likelihood that both PORVs will actuate simultaneously during credible OPPS events. Similarly, WCAP 14040-NP-A indicates that when there is insufficient range between the upper and lower pressure limits to select PORV setpoints that provide protection against violating both limits, then the setpoint selection that provides protection against the upper limit violation takes precedence. WCAP-14040-NP Revision 1 was approved by the NRC by letter dated October 16, 1995, which was incorporated in Revision 2 of the approved WCAP issued in January 1996.

Modification of the heatup and cooldown curves and OPPS setpoints does not alter any assumptions previously made in the radiological consequence evaluations nor affect mitigation of the radiological consequences of an accident described in the Updated Final Safety Analysis Report (UFSAR). Therefore, the proposed changes will not significantly increase the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed heatup and cooldown curves applicable for the first 15 EFPY were generated using approved methodology and Code Case N-640. Generating these curves with Code Case N-640 reduced the excess conservatism that exists in the current curves and results in an increase in the safety of the plant, as the likelihood of RCP seal failures and/or fuel problems will decrease. The change does not cause the initiation of any accident nor create any new single failure.

   The modification of the OPPS setpoints ensures that the RCPB integrity is protected at low temperatures. The new setpoints were selected using conservative assumptions to ensure that sufficient margin is available to prevent violation of the P/T limits due to anticipated mass and heat input transients. The modification of the setpoints does not change, degrade, or prevent the safe response of the RCS to accident scenarios, as described in UFSAR Chapter 15. The proposed change does not cause the initiation of any accident nor create any new credible single failure.

   Therefore, the proposed license amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

   The new P/T curves define the limits for ensuring prevention of nonductile failure for the reactor vessel, and does not significantly reduce the margin of safety for the plant. The methodology provided in Code Case N-640 removed some of the excess conservatism from the current Appendix G analysis. However, this improved overall plant safety by expanding the operating window relative to the RCP seal requirements. The probability of damaging the RCP seals is reduced. Therefore, the margin of safety is not significantly reduced.

   The OPPS setpoints will continue to ensure the RCS pressure boundary will be protected from pressure transients. They were generated using the proposed heatup and cooldown curves as input. The OPPS setpoints include additional margin by including instrument uncertainties not included in the current setpoints. Therefore, the margin of safety is not significantly reduced.

F. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Based on the considerations expressed above, it is concluded that the activities associated with this license amendment request satisfy the requirements of 10 CFR 50.92(c) and, accordingly, a no significant hazards consideration finding is justified.

G. ENVIRONMENTAL CONSIDERATION

This license amendment request changes 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. It has been determined that this license amendment request involves 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. This license amendment request may change requirements with respect to installation or use of a facility component located within the restricted area or change an inspection or surveillance requirement; however, the category of this licensing action does not individually or cumulatively have a significant effect on the human environment. Accordingly, this license amendment request meets 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 this license amendment request.

H. UFSAR CHANGES

No UFSAR changes to implement this License Amendment Request are required.

REFERENCES

1. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.

2. WCAP 14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," J. D. Andracheck et al, January 1996.

3. WCAP-9615, Revision 1, "Duquesne Light Company Beaver Valley Unit 2 Reactor Vessel Radiation Surveillance Program," P. A. Peter, June 1995.

4. Section XI of the ASME Boiler and Pressure Vessel Code, Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

5. WCAP-14485, "Beaver Valley Unit 2 Heatup and Cooldown Curves for Normal Operation," P. A. Grendys, March 1996.

6. WCAP-15139, "Beaver Valley Unit 2 Heatup and Cooldown Limit Curves During Normal Operation at 15 EFPY Using Code Case N-626," T. J. Laubham, January 1999.

7. EDRE-SMT-98-135, "Technical Basis for Revised P-T Limit Curve Methodology," W. H. Bamford et al, December 1998.

8. Code of Federal Regulations, 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements," U.S. Nuclear Regulatory Commission, Washington, D. C., Federal Register, Volume 60, No. 243, dated December 19, 1995.

9. NPD-OPES(99)-055, "Low Temperature Overpressure Protection System Setpoint Review for Beaver Valley Unit 2 15 EFPY Heatup and Cooldown Curves," March 1999.

ATTACHMENT C

Beaver Valley Power Station, Unit No. 2
License Amendment Request No. 127
UPDATE HEATUP AND COOLDOWN CURVES AND OVERPRESSURE
PROTECTION SYSTEM (OPPS) SETPOINT CURVE

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WCAP-15139
Beaver Valley Unit 2 Heatup and Cooldown Limit Curves During Normal
Operation at 15 EFPY Using Code Case N-626 (January 1999)