FENOC

FirstEnergy Nuclear Operating Company

Beaver Valley Power Station
P.O. Box 4
Shippingport, PA 15077-0004

Lew W. Myers
Senior Vice President

724-682-5234
Fax: 724-643-8069

June 29, 2001
L-01-087

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
License Amendment Request No. 292

Pursuant to 10 CFR 50.90, FirstEnergy Nuclear Operating Company (FENOC) requests an amendment to the above license in the form of changes to the technical specifications. The proposed amendment would update the technical specification heatup and cooldown curves, and the overpressure protection setpoints and Bases to apply for up to 22 effective full power years (EFPY). The applicability for the current curves will expire at 16 EFPY. Therefore, approval of this license amendment is requested by February 1, 2002, to implement the amendment and allow continued operation beyond this time frame.

The proposed amendment also incorporates changes contained in approved Technical Specification Traveler Forms (TSTF) applicable to Beaver Valley Power Station (BVPS) Unit 1 Technical Specification 3/4.4.9.3, "Overpressure Protection Systems." As a result, Technical Specification 3/4.4.9.3 and its associated Bases are revised to reflect the applicable changes.

Proposed technical specification changes are presented in Attachment A. The safety analysis (including the no significant hazards evaluation) is presented in Attachment B. An exemption request from 10 CFR 50, Appendix G and ASME Section XI, Appendix G requirements for reactor vessel pressure limits at low temperatures is presented in Attachment C. A proprietary (Class 2C) and non-proprietary (Class 3) version of Westinghouse report "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Y-Capsule," Revision 1, dated April 2001 and May 2001, respectively, is included in Attachment D. WCAP-15570, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," Revision 2, April 2001, is included in Attachment E. The Proprietary Information Notice, Copyright Notice, a Westinghouse application for withholding proprietary information (CAW-01-1457), applicable to the Class 2C report, is provided in Attachment F.

As the Class 2C report contains information proprietary to Westinghouse Electric Company, it is supported by an affidavit signed by Westinghouse, the owner of the

AP01

information. This affidavit sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.790 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to the copyright or proprietary aspects of the Class 2C report or the supporting Westinghouse Affidavit should reference CAW-01-1457 and should be addressed to H. A. Sepp, Regulatory and Licensing Engineering, Westinghouse Electric Company, LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

This change has been reviewed by the Beaver Valley review committees. The change was determined to be safe and does not involve a significant hazard consideration as defined in 10 CFR 50.92 based on the attached safety analysis and no significant hazard evaluation. An implementation period of up to 60 days is requested following the effective date of this amendment.

If there are any questions concerning this matter, please contact Mr. Thomas S. Cosgrove, Manager, Regulatory Affairs at 724-682-5203.

Sincerely,

Lew W. Myers

c: Mr. L. J. Burkhart, Project Manager
Mr. D. M. Kern, Sr. Resident Inspector
Mr. H. J. Miller, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

Subject: **Beaver Valley Power Station, Unit No. 1**
BV-1 Docket No. 50-334, License No. DPR-66
License Amendment Request No. 292

I, Marc P. Pearson, being duly sworn, state that I am Director, Nuclear Services of FirstEnergy Nuclear Operating Company (FENOC), that I am authorized to sign and file this submittal with the Nuclear Regulatory Commission on behalf of FENOC, and that the statements made and the matters set forth herein pertaining to FENOC are true and correct to the best of my knowledge and belief.

FirstEnergy Nuclear Operating Company

Marc P. Pearson

Marc P. Pearson
Director, Nuclear Services - FENOC

COMMONWEALTH OF PENNSYLVANIA

COUNTY OF BEAVER

Subscribed and sworn to me, a Notary Public, in and for the County and State above named, this 29th day of June, 2001.

Tracey A. Baczek

My Commission Expires:

Notarial Seal
Tracey A. Baczek, Notary Public
Shippingport Boro, Beaver County
My Commission Expires Aug. 16, 2001
Member, Pennsylvania Association of Notaries

ATTACHMENT A

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 292

The following is a list of the affected pages:

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Figure Index

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

(Proposed Wording)

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop (A) and its associated steam generator and reactor coolant pump,#

2. Reactor Coolant Loop (B) and its associated steam generator and reactor coolant pump,#

3. Reactor Coolant Loop (C) and its associated steam generator and reactor coolant pump,#

4. Residual Heat Removal Pump (A) and a heat exchanger,**

5. Residual Heat Removal Pump (B) and a second heat exchanger.**

b. At least one of the above coolant loops shall be in operation.***

APPLICABILITY: Modes 4 AND 5.

** The normal or emergency power source may be inoperable in MODE 5.

*** All reactor coolant pumps and Residual Heat Removal pumps may be de-energized for up to 1 hour provided: 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration and 2) core outlet temperature is maintained at least 10°F below saturation temperature. For purposes of this specification, the addition of borated water to the RCS does not constitute dilution of the RCS boron concentration provided the boron concentration of the borated water being added is greater than the minimum required to satisfy the requirements of Specification 3.1.1.1 for Mode 4; or Specification 3.1.1.2 for Mode 5.

The first

No reactor coolant pump in a non-isolated loop shall be started with one or more non-isolated RCS cold leg temperatures less than or equal to the enable temperature set forth in Specification 3.4.9.3, unless the secondary side water temperature of each steam generator in a non-isolated loop is less than 25°F above each of the non-isolated RCS cold leg temperatures.

not

50°F

(Proposed Wording)

(Replace with Revised Figure 3.4-2)

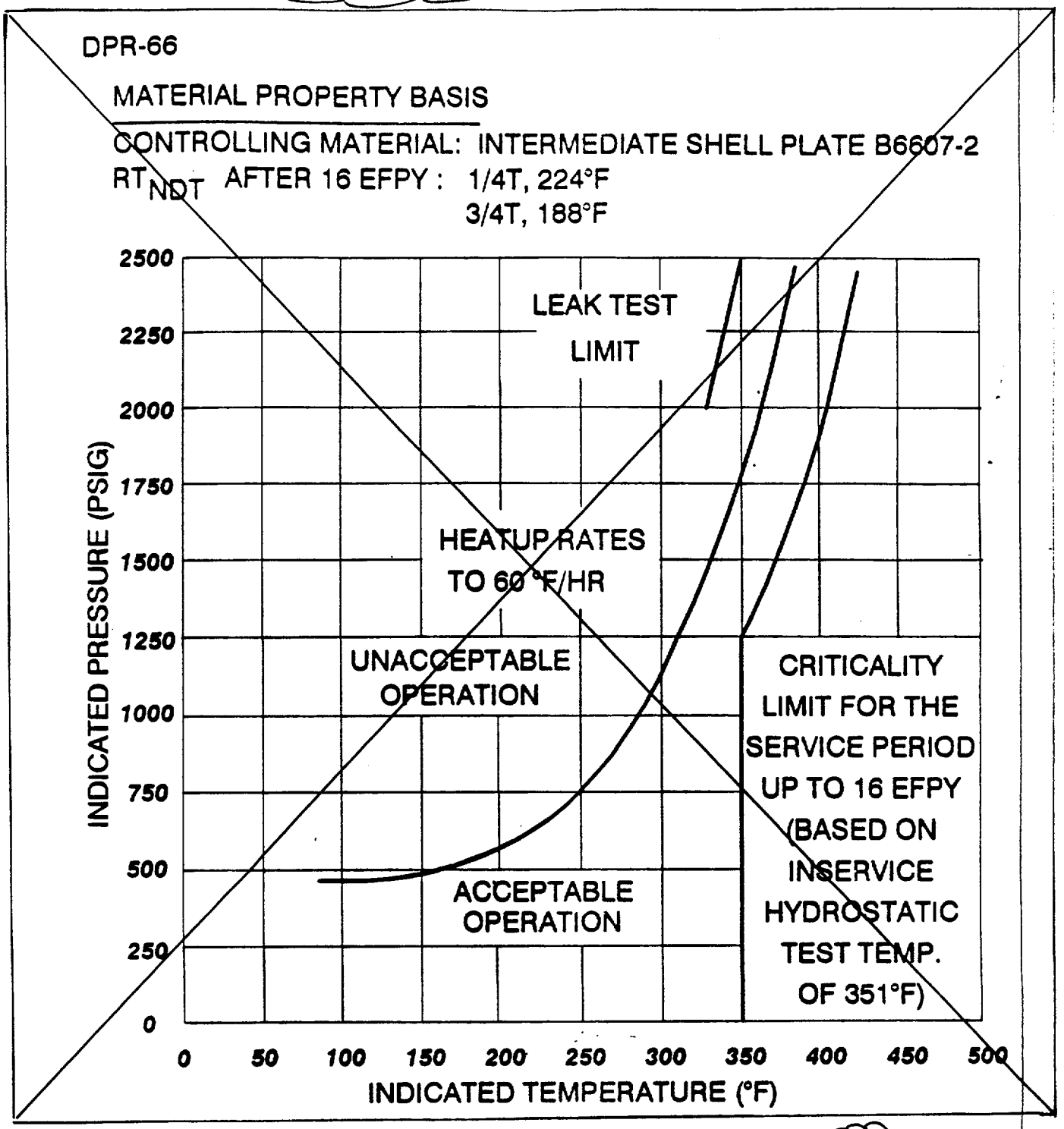

FIGURE 3.4-2 (22)

Beaver Valley Unit 1 Reactor Coolant System Heatup Limitations Applicable for the First ~~16~~ 22 EFPY

(Proposed Wording)

DPR-66

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL PLATE

LIMITING ART VALUES AT 22 EFPY: 1/4T, 233°F

3/4T, 196°F

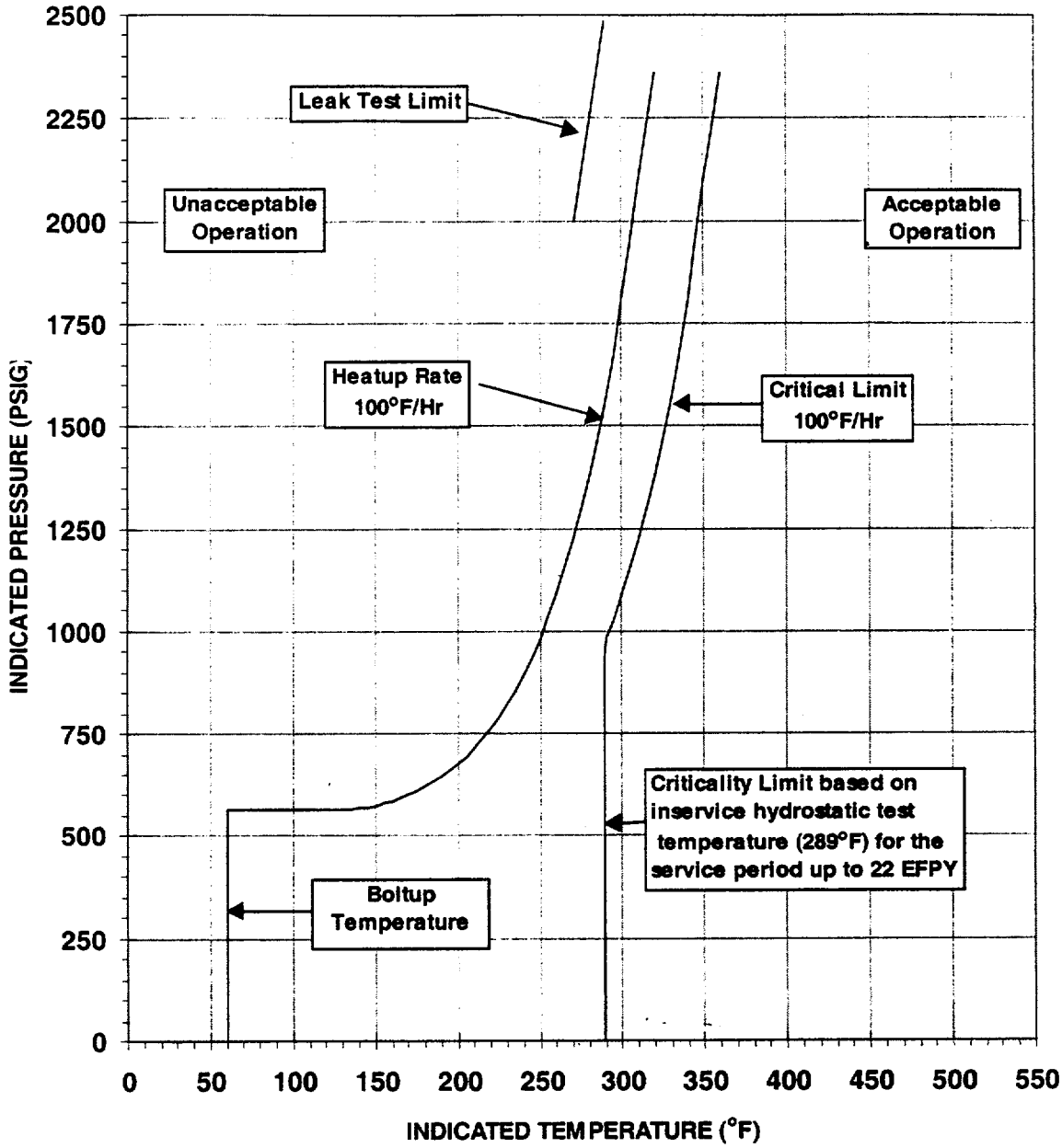

FIGURE 3.4-2

Beaver Valley Unit 1 Reactor Coolant System Heatup

Limitations Applicable for the First 22 EFPY

(Proposed Wording)

(Replace with Revised Figure 3.4-3)

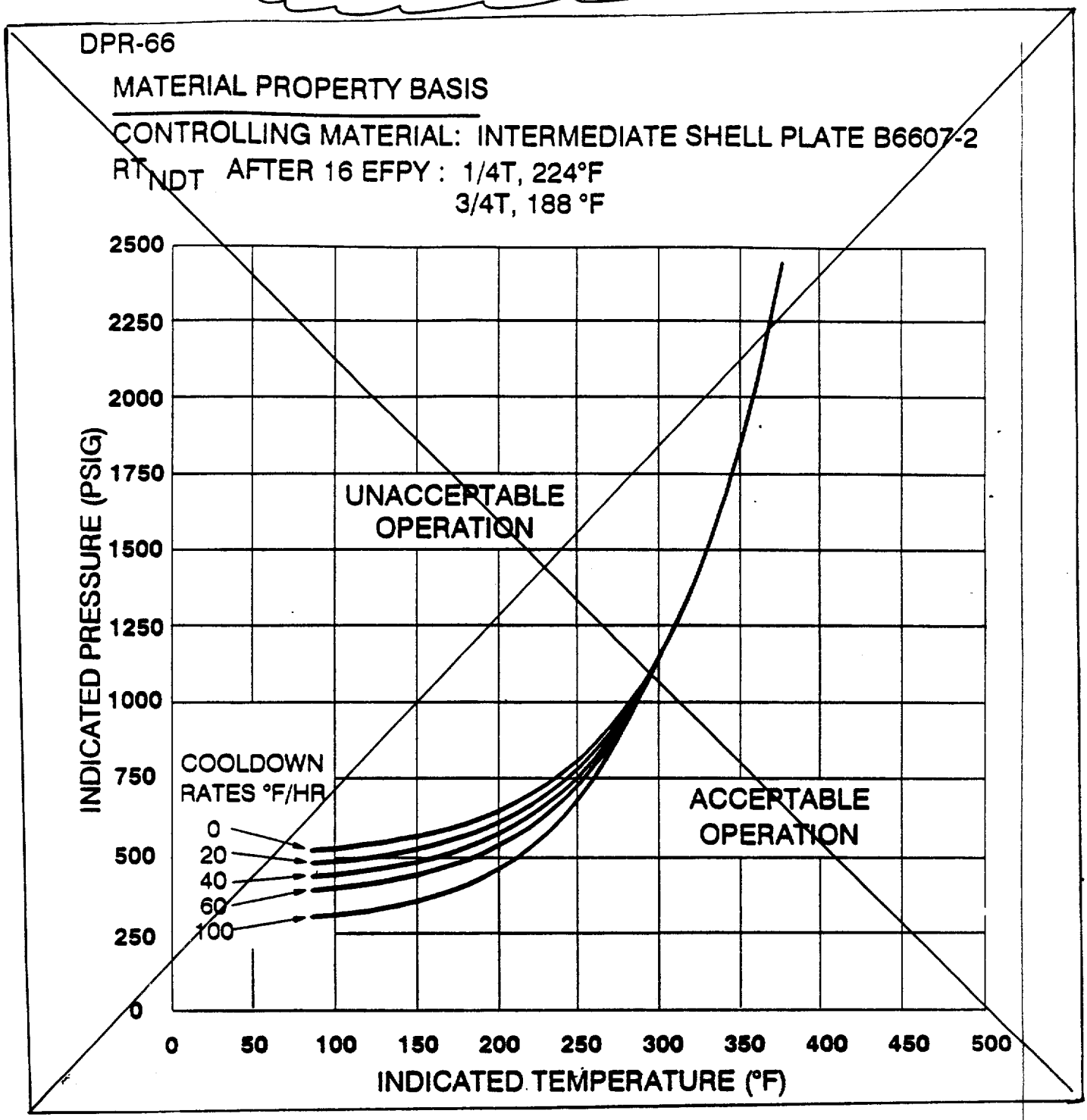

FIGURE 3.4-3

Beaver Valley Unit 1 Reactor Coolant System Cooldown Limitations Applicable for the First ~~16~~ 22 EFPY

(NEXT PAGE IS 3/4 4-27)

(Proposed Wording)

DPR-66

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE & LOWER SHELL PLATE

LIMITING ART VALUES AT 22 EFPY: 1/4T, 233°F
3/4T, 196°F

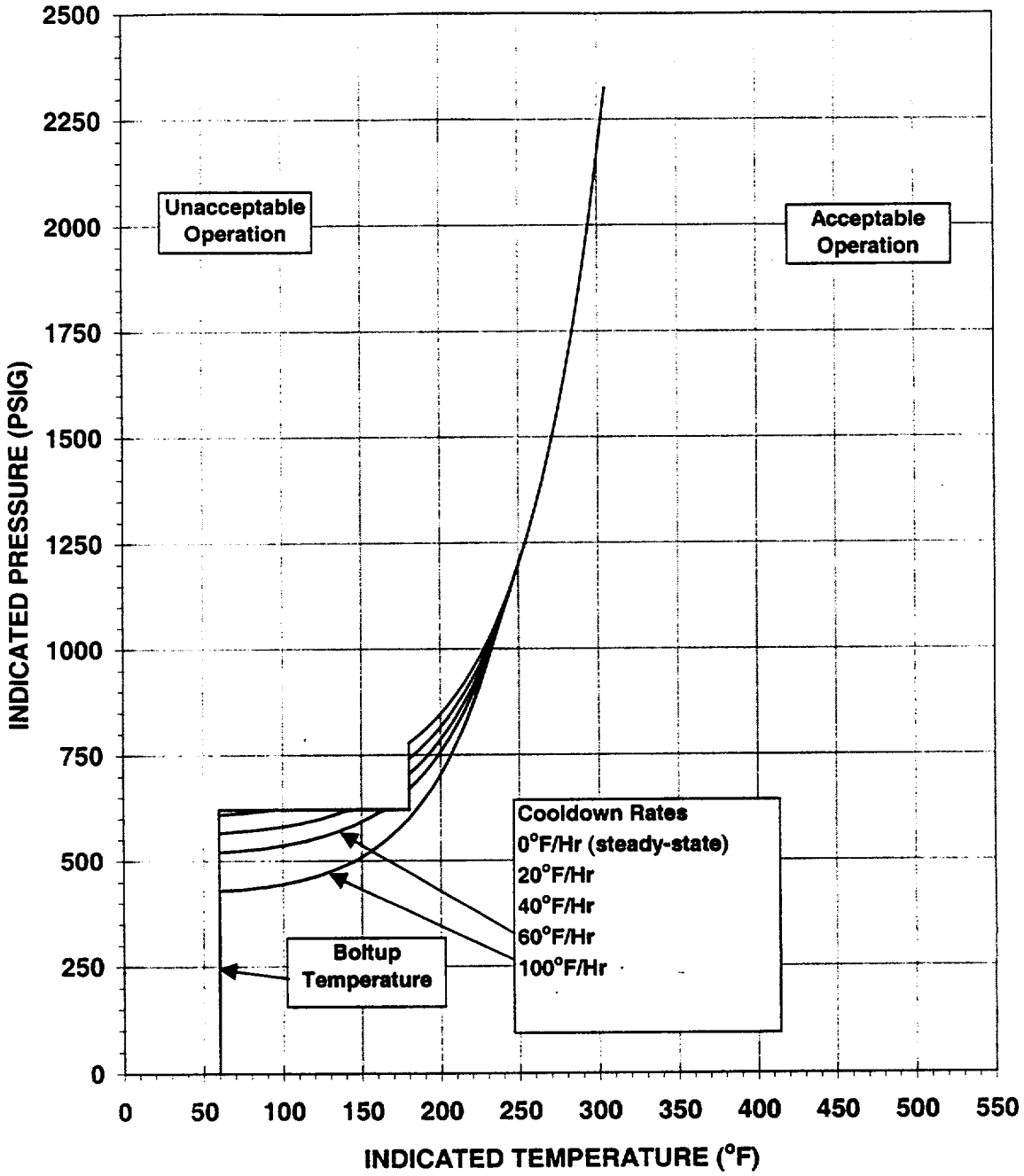

FIGURE 3.4-3
Beaver Valley Unit 1 Reactor Coolant System Cooldown
Limitations Applicable for the First 22 EFPY

(Proposed Wording)

OVERPRESSURE PROTECTION SYSTEMS

(NOMINAL / MAXIMUM)

LIMITING CONDITION FOR OPERATION

3.4.9.3 An overpressure protection system shall be OPERABLE with a maximum of one charging pump(1) capable of injecting into the RCS and the accumulators isolated(2) and either a or b below:

a. Two power operated relief valves (PORVs) with a lift setting less than or equal to ~~432~~ 403 psig, or

b. The RCS depressurized and an RCS vent of greater than or equal to 2.07 square inches.

APPLICABILITY: Mode 4 when any RCS cold leg temperature is less than or equal to an enable temperature of ~~329~~ 343°F,
Mode 5,
Mode 6 when the reactor vessel head is on.

ACTION:

a. With two or more charging pumps capable of injecting into the RCS, immediately initiate action to verify a maximum of one charging pump is capable of injecting into the RCS or depressurize and vent the RCS through a 2.07 square inch or larger vent within 12 hours.

b. With an accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the heatup and cooldown curves, isolate the affected accumulator within 1 hour or increase the RCS cold leg temperature above the enable temperature within the next 12 hours or depressurize the affected accumulator to less than the maximum RCS pressure for the existing cold leg temperature allowed by the heatup and cooldown curves within the next 12 hours.

c. With one PORV inoperable in MODE 4 (when any RCS cold leg temperature is less than or equal to the enable temperature), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a 2.07 square inch or larger vent within the next 12 hours.

(1 hour)

(1) Two charging pumps may be capable of injecting into the RCS for pump swap operation for less than or equal to 1~~5 minutes~~.

(2) Accumulator isolation with power removed from the discharge isolation valves is only required when the accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the heatup and cooldown curves.

(Proposed Wording)

BASES

3/4.4.1.1, 2, 3 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR limit during all normal operations and anticipated transients. In Modes 1 and 2, with one reactor coolant loop not in operation, THERMAL POWER is restricted to less than or equal to 31 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above the design DNBR limit. A loss of flow in two loops will cause a reactor trip if operating above P-7 (11 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (31 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, due to the initial conditions assumed in the analysis for the control rod bank withdrawal from a subcritical condition, two operating coolant loops are required to meet the DNB design basis for this Condition II event.

In MODES 4 and 5, a single reactor coolant loop or RHR subsystem provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more non-isolated RCS cold legs less than or equal to the enable temperature set forth in Specification 3.4.9.3 are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary side water temperature of each steam generator in a non-isolated loop is less than ~~25°F~~ 50°F above each of the non-isolated RCS cold leg temperatures. The secondary side water temperature is to be verified by direct measurements of the fluid temperature, or contact temperature readings on the steam generator secondary, or blowdown piping after purging of stagnant water within the piping. This shall be determined within 10 minutes prior to starting ~~a~~ the first reactor coolant pump.

(Proposed Wording)

3/4.4.8 SPECIFIC ACTIVITY (Continued)

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 0.35 µCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 0.35 µCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limits shown on Figure 3.4-1 must be restricted to ensure that assumptions made in the UFSAR accident analyses are not exceeded.

Reducing Tavg to < 500°F minimizes the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. This action also reduces the pressure differential across the steam generator tubes reducing the probability and magnitude of main steam line break accident induced primary-to-secondary leakage. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal-induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

(Proposed Wording)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS(Continued)

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

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~~ 100°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 time in life indicated on the respective curves.

The reactor vessel materials have been tested to determine their initial RT_{NTD}; 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_{NTD}. Therefore, an adjusted reference temperature, based upon the fluence ~~and~~ copper content and Nickel content of the material in question, can be predicted using ~~Figures B 3/4.4-1 and B 3/4.4-2~~ WCAP-15570, Rev. 2 and Regulatory Guide 1.99, Revision 2. The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NTD}.

The heatup and cooldown curves have been developed in accordance with the methodology provided in Regulatory Guide 1.99, Revision 2 and no longer contain the additional margin of 10°F and 60 psig for instrument error previously incorporated in these curves.

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DPR-66

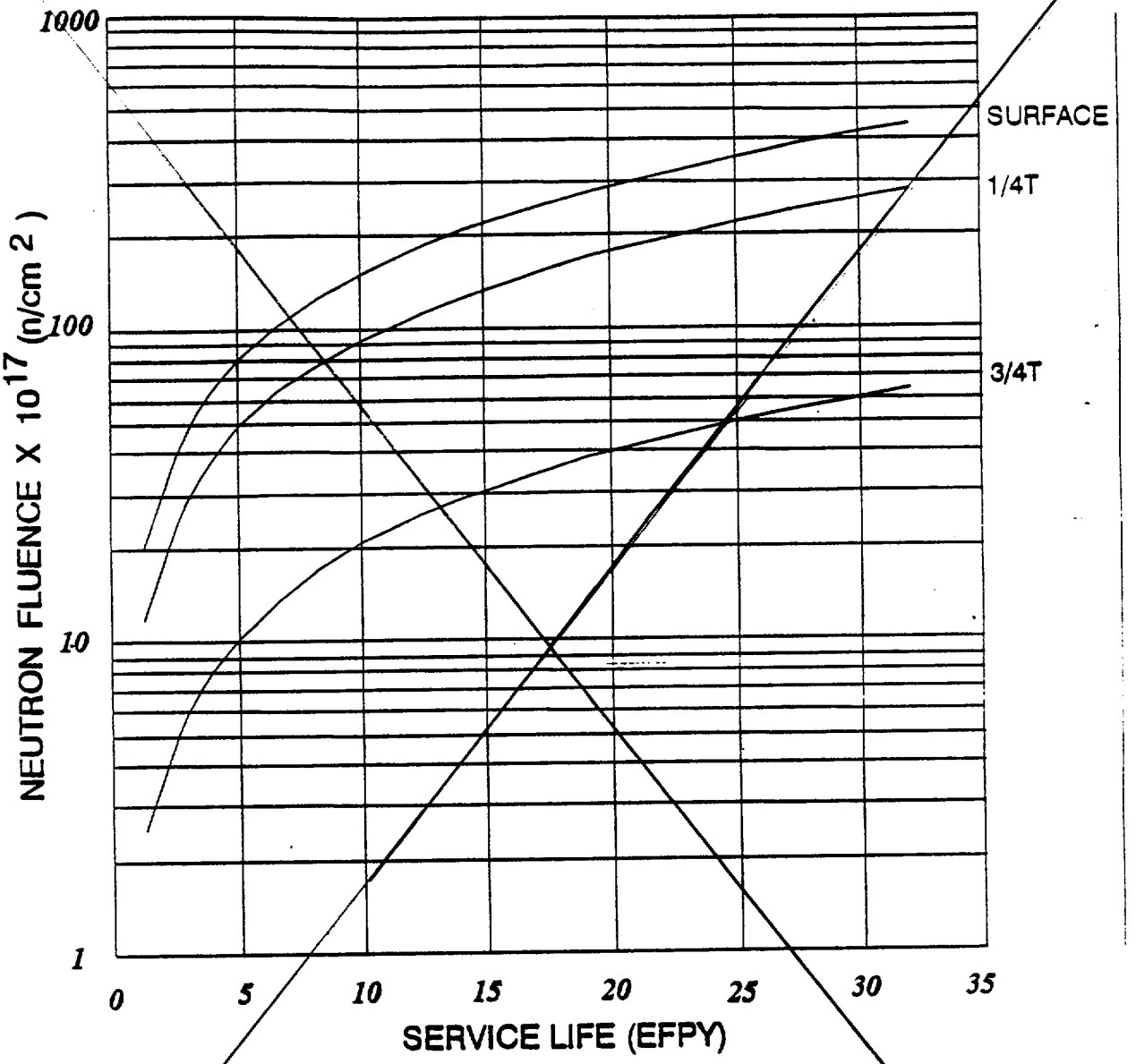

FIGURE B 3/4 4-1

FAST NEUTRON FLUENCE (E>1 Mev) AS A FUNCTION OF FULL POWER SERVICE LIFE (EFPY)

(Replace with Insert 1)

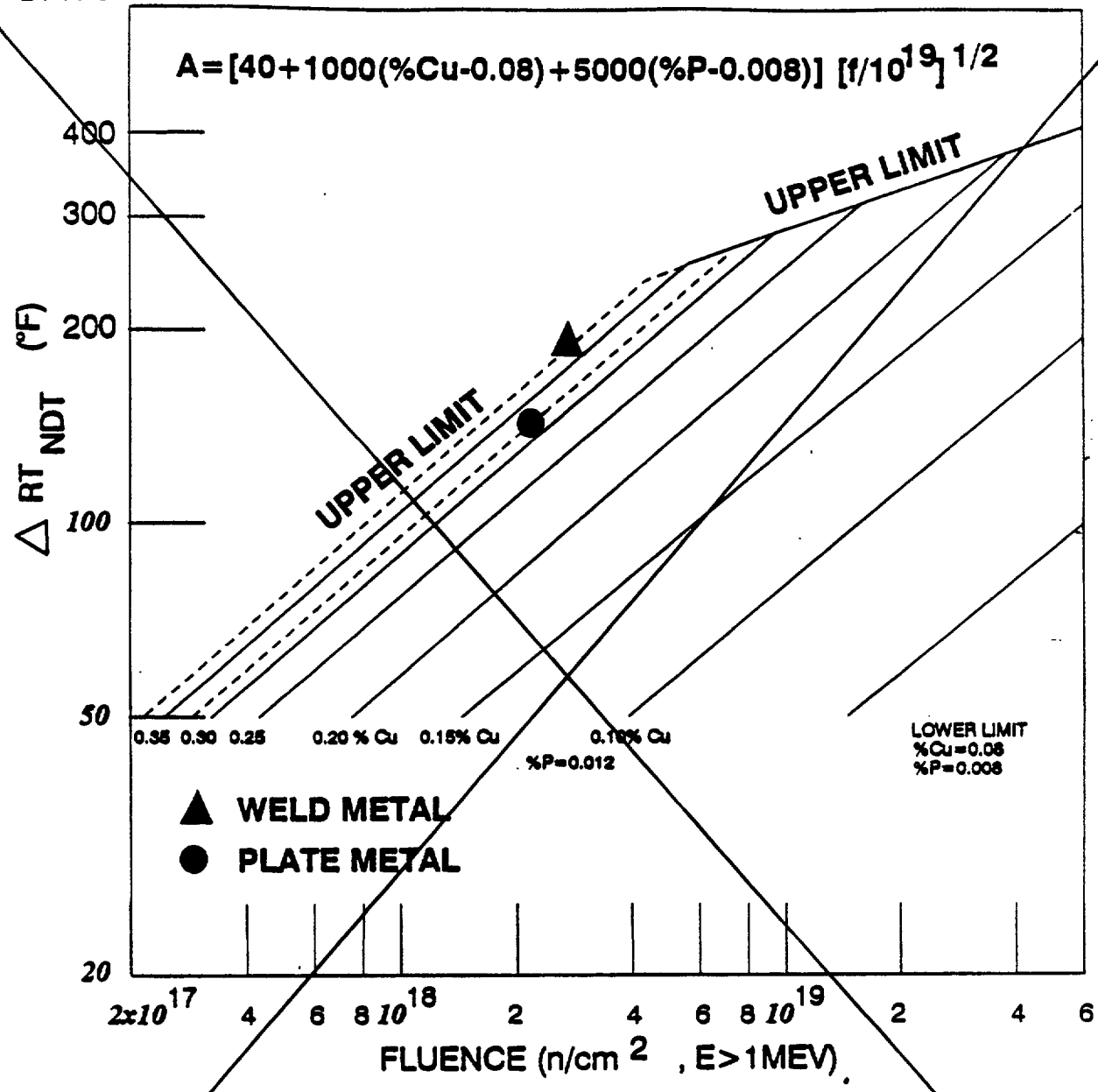

FIGURE B 3/4 4-2

EFFECT OF FLUENCE, COPPER CONTENT, AND PHOSPHORUS CONTENT ON Δ RT_{NDT} FOR REACTOR VESSEL STEELS PER REGULATORY GUIDE 1.99

(Proposed Wording)

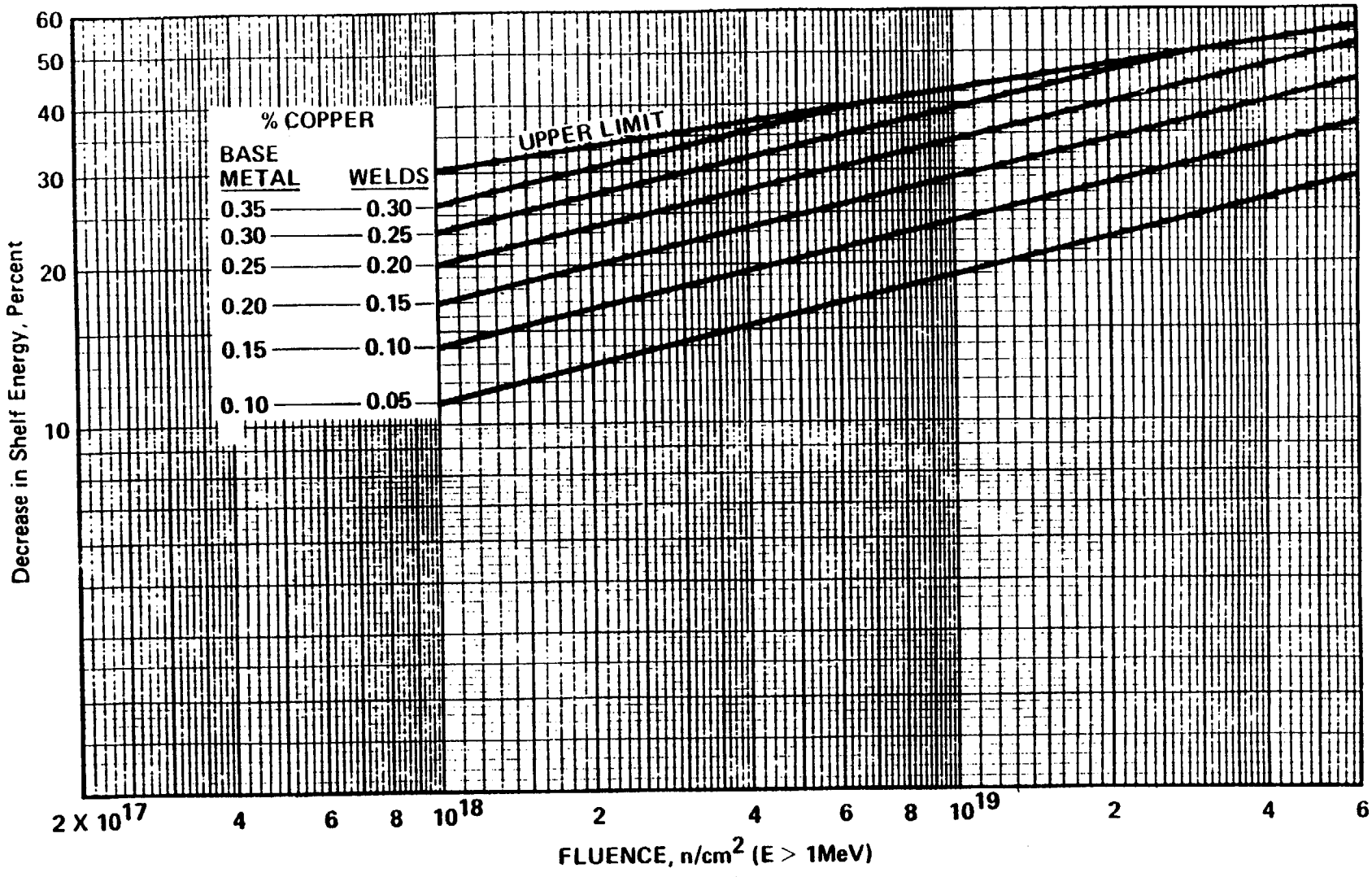

FIGURE B 3/4 4-1
PREDICTED DECREASE IN SHELF ENERGY AS A FUNCTION OF COPPER CONTENT AND FLUENCE

Insert 1

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Ni (%)

Component	Heat No.	Code No.	Material Type	Cu (%)	Ni (%)	P (%)	T_{NDT} (°F)	RT_{NDT} (°F)	Upper Shelf Energy (Ft-lb) MWD	Upper Shelf Energy (Ft-lb) NMWD
Closure Head Dome	C6213-1B	B6610	A533B CL. 1	.15	—	.010	-40	0*	121	—
Closure Head Seg.	A5518-2	B6611	A533B CL. 1	.14	—	.015	-20	-20*	131	—
Closure Head Flange	ZV3758	—	A508 CL. 2	.08	—	.007	60*	60*	>100	—
Vessel Flange	ZV3661	—	A508 CL. 2	.12	—	.010	60*	60*	166	—
Inlet Nozzle	9-5443	—	A508 CL. 2	.10	—	.008	60*	60*	82.5	—
Inlet Nozzle	9-5460	—	A508 CL. 2	.10	—	.010	60*	60*	94	—
Inlet Nozzle	9-5712	—	A508 CL. 2	.08	—	.007	60*	60*	97	—
Outlet Nozzle	9-5415	—	A508 CL. 2	—	—	.008	60*	60*	97	—
Outlet Nozzle	9-5415	—	A508 CL. 2	—	—	.007	60*	60*	112.5	—
Outlet Nozzle	9-5444	—	A508 CL. 2	.09	—	.007	60*	60*	103	—
Upper Shell	123V339	—	A508 CL. 2	—	—	.010	40	40*	155	—
Inter Shell	C4381-2	B6607-2	A533B CL. 1	.14	.62	.015	-10	73	123	82.5
Inter Shell	C4381-1	B6607-1	A533B CL. 1	.14	.62	.015	-10	43	128.5	90
Lower Shell	C6317-1	B6903-1	A533B CL. 1	.20	.54	.010	-50	27	134	60
Lower Shell	C6293-2	B7203-2	A533B CL. 1	.14	.57	.015	-20	20	129.5	83.5
Trans Ring	123V223	—	A508 CL. 2	—	—	—	30	30*	143	—
Botttom Hd Seg	C4423-3	B6618	A533B CL. 1	.13	—	.008	-30	-29*	124	—
Bottom Hd Dome	C4482-1	B6619	A533B CL. 1	.13	—	.015	-50	-33*	125.5	—
~~Core Region Welds~~				~~.30-.37~~		~~.013~~		~~0*~~		~~>100~~
Weld HAZ				—	—	—	-40	-40	—	136.5

Insert 2

*Estimated Per NRC Standard Review Plan Branch Technical Position MTEB 5-2

MWD - Major Working Direction

NMWD - Normal to Major Working Direction

Insert 3

(Proposed Wording)

INSERT 2

Inter to Lower Shell Weld	90136	---	----	.27	.07	---	---	-56	---	>100
Inter Shell Long. Weld	305424	---	----	.28	.63	---	---	-56	---	>100
Lower Shell Long. Weld	305414	---	----	.34	.61	---	---	-56	---	>100

INSERT 3

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

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 ΔRT_{NDT}. RT_{NDT} is designated as the higher of either the drop weight nil-ductility 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, 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 ~~phosphorus~~) present in reactor vessel steels. The Regulatory Guide 1.99, ~~trend~~ curves, which show the effect of fluence and copper ~~and phosphorus~~ contents on ΔRT_{NDT} for reactor vessel steels are shown in Figure B 3/4.4-~~2~~. 1

upper shelf energy (USE)

nickel

Given the copper and ~~phosphorus~~ contents of the most limiting material, the radiation-induced ΔRT_{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}.

can be generated

Revision 2

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

where f = fluence and CF = chemistry factor, a function of copper and nickel.

(Proposed Wording)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The preirradiation fracture-toughness properties of the Beaver Valley Unit 1 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 NRC Regulatory Standard Review plan.[1] The postirradiation fracture toughness properties of the reactor vessel beltline material were obtained directly from the Beaver Valley Unit 1 Vessel Material Surveillance Program. Insert 4

Insert 5

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:

IC

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

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:

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

(Code Case N-640)

1. "Fracture Toughness Requirements," Branch Technical Position MTEB No. 5-2, Section 5.3.2-14 in Standard Review Plan, NUREG-75/087, 1975.

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. 461-469, 1980 Edition, American Society of Mechanical Engineers, New York, 1980.

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

(Proposed Wording)

INSERT 4

, determined in accordance with 1996 Addenda to ASME Section XI, Appendix G and ASME Code Case N-640,

INSERT 5

Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (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. Adjusted Reference Temperature (ART), defined as ART = initial RT_{NDT} + Margins for uncertainties + ΔRT_{NDT}, is used to index the material to the K_{IC} 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. The most limiting ART values are used in the generation of heatup and cooldown pressure-temperature limit curves for normal operation.

The pressure-temperature limit curves are developed using ASME 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 ASME 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.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

where:

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

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 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.

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 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 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.

(Proposed Wording)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The above procedures are needed because there is no direct control on temperature at the 1/4T 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 1/4T 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} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature.

IC

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 1/4T 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.

IC

(Proposed Wording)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T 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. (As documented in WCAP-15570, Revision 2)

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.

The actual shift in ~~NDTT~~ 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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

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 ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

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 6)

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in UFSAR Table 4.5-3 to assure compliance with the requirements of Appendix H to 10 CFR 50.

The limitations imposed on the pressurizer heatup and cooldown rates and 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 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 ASME 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 (including ASME Code CASE N-640) requirements during the OPPS MODES.

(Proposed Wording)

INSERT 6

These pressure-temperature limits lines on Figures 3.4-2 and 3.4-3 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.

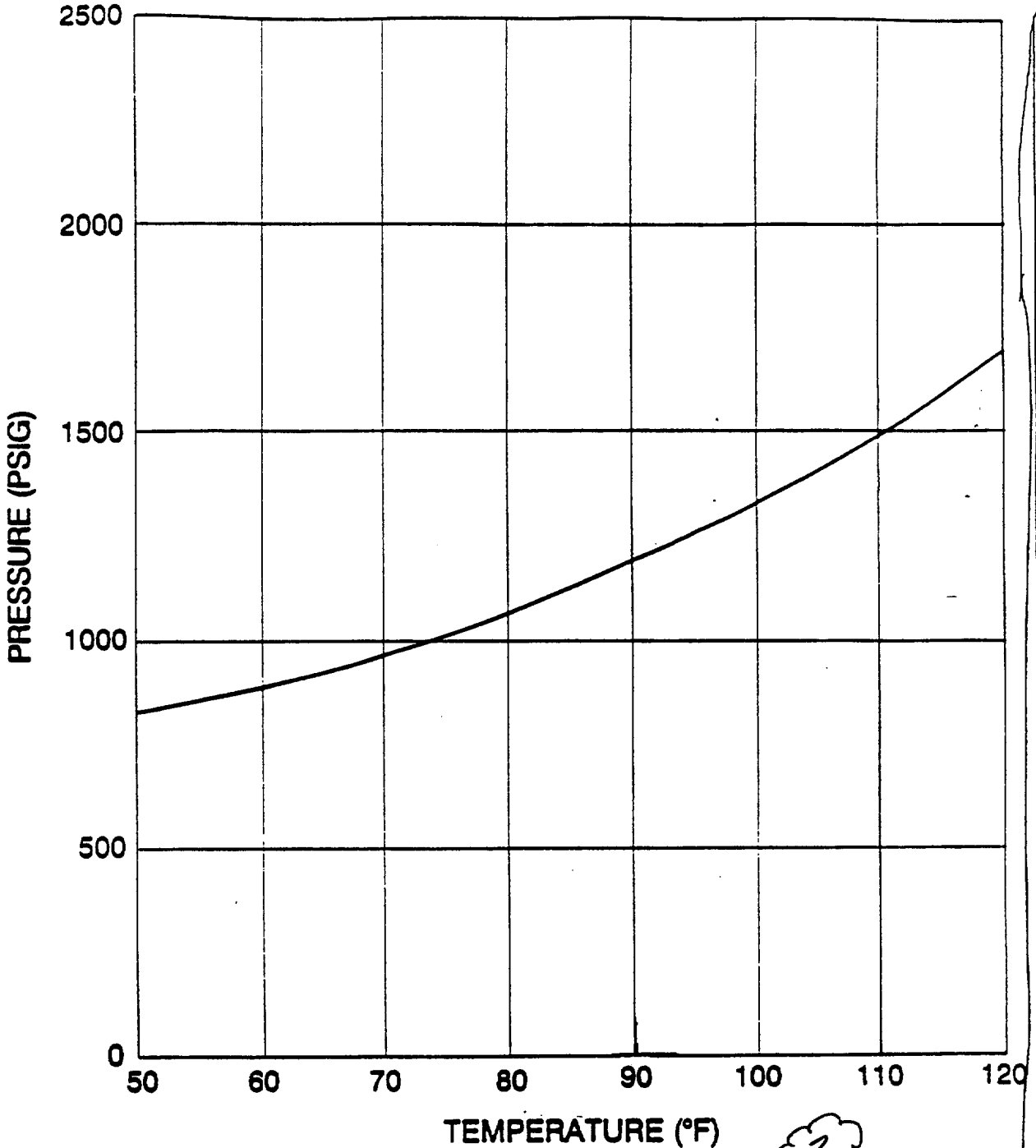

FIGURE B 3/4 4-3 (2)

ISOLATED LOOP PRESSURE-TEMPERATURE LIMIT CURVE

(Proposed Wording)

~~Added by NRC letter dated 3/2/92~~

BASES (Continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

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 circuit. The OPPS actuation circuit monitors RCS pressure and determines when a condition not acceptable is approached. If the indicated pressure meets or exceeds the OPPS actuation setpoint, 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.

(Insert 7)

RCS VENT REQUIREMENTS

Once the RCS is depressurized, a vent exposed to the pressurizer relief tank (PRT) or containment atmosphere will maintain the RCS 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 PRT or containment atmosphere of the required size through any positive means available which cannot be inadvertently defeated. The vent 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.

(Proposed Wording)

INSERT 7

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.

(including ASME Code Case N-640)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

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.

The heatup and cooldown curves represent the Appendix G limits that define OPPS operation. Setpoint calculations correlated to RCS temperature define acceptable OPPS setpoints for steady-state pressure-temperature limits based on Revision 2 of NRC Regulatory Guide 1.99. Any change to the RCS that may affect OPPS operation must be evaluated against the analyses to determine the impact of the change on the OPPS acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

MASS INPUT TYPE TRANSIENTS

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

HEAT INPUT TYPE TRANSIENTS

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the OPPS MODES to ensure that mass and heat input transients do not occur, which either of the OPPS overpressure protection means cannot handle:

a. Deactivating all but one charging pump OPERABLE;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

, except during pump swapping operations as addressed in the LCO

BASES (Continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

(Insert 8)

HEAT INPUT TYPE TRANSIENTS (Continued)

c. ~~Disallowing start of an RCP if the secondary side water temperature of each steam generator in a non-isolated loop is greater than or equal to 25°F above the non-isolated RCS cold leg temperature in any non-isolated loop. LCO 3.4.1.3, "Reactor Coolant System - Shutdown," provides this protection.~~

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 OPPS 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.

(Insert 9)

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.

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 2.07 square inches is capable of mitigating the allowed OPPS overpressure transient. The capacity of

(Proposed Wording)

INSERT 8

Meeting the secondary side water to RCS cold leg temperature difference requirement specified in LCO 3.4.1.3, "Reactor Coolant System - Shutdown."

INSERT 9

The Maximum Allowable Nominal 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 allowable nominal setpoint ensures that 10 CFR 50 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 reactor vessel and the RCS wide range temperature indicator used for OPPS; (3) instrument uncertainties; (4) single failure; and (5) the pressure difference between the wide range pressure transmitter and the reactor vessel limiting beltline region.

BASES (Continued)

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

RCS VENT PERFORMANCE (Continued)

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 analysis.

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.

(be) (that a maximum of)

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.

(1 hour)

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.

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. Two OPERABLE PORVs; a PORV is OPERABLE for OPPS when its block valve is open, its lift setpoint is set to the limit and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits; or

b. A depressurized RCS and an RCS vent.

(Proposed wording)

ATTACHMENT B

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 292
REVISION OF UNIT NO. 1 PRESSURE AND TEMPERATURE LIMITS

A. DESCRIPTION OF AMENDMENT REQUEST

The proposed license amendment is applicable to Beaver Valley Power Station (BVPS) Unit No. 1. It revises the reactor coolant system (RCS) pressure and temperature (P/T) limits to be valid through 22 effective full power years (EFPY). It also revises the overpressure protection system (OPPS) power operated relief valves (PORVs) setpoint and the OPPS enabling temperature. These changes have been prepared using the NRC-approved methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996, with two exceptions. They include the use of:

a) ASME Code Case N-640, "Alternate Reference Fracture Toughness for Development of P-T Curves for Section XI, Division 1," March 1999, and

b) ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," December 1995, (through 1996 Addendum).

Pursuant to 10 CFR 50.12(a)(2)(iii), FirstEnergy Nuclear Operating Company (FENOC) requests exemption to 10 CFR 50, Appendix G, based on American Society of Mechanical Engineers (ASME) Code Case N-640. The exemption request is included as Attachment C. The proposed Unit No. 1 P/T limits incorporate the results from testing of Capsule Y described in WCAP-15571, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," Revision 0, November 2000.

The following provides a summary of the proposed changes.

1. Technical Specification 3/4.4.1.3, "Reactor Coolant System – Shutdown," and the associated Bases are revised by changing the secondary to primary temperature difference limit restriction associated with starting an idle reactor coolant pump (RCP) from 25°F to 50°F.

2. Technical Specification 3/4.4.9.1, "Reactor Coolant System Pressure/Temperature Limits," is revised in accordance with the NRC-approved methodology described in WCAP-14040-NP-A, Revision 2. The revision changes the effective full power years (EFPY) from 16 to 22 and changes the heatup and cooldown curves shown on Figures 3.4-2 and 3.4-3 appropriately. The heatup and cooldown curves are revised in accordance with WCAP-15570, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," Revision 2, April 2001. The revised heatup curve, Figure 3.4-2, changes the heatup curve from 60°F/hour to 100°F/hour. This change is discussed in Section C of this license amendment request.

3. Technical Specification 3/4.4.9.3, "Reactor Coolant System Overpressure Protection Systems," is revised in accordance with the NRC-approved methodology described in WCAP-14040-NP-A, Revision 2. The revision changes the PORV lift setting from 432 psig to 403 psig and the OPPS enable temperature from 329°F to 343°F. These changes are documented in Reference 6, "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Y-Capsule," Revision 1, May 2001. This report is included as Attachment D.

   A review of approved Technical Specification Traveler Forms (TSTF) pertaining to Technical Specification 3/4.4.9.3 was conducted for BVPS Unit No. 1 applicability. As a result of this review, a portion of TSTF-285, Rev. 1, "Charging Pump Swap LTOP Allowance," has been incorporated into this license amendment request. The portion incorporated consists of Note 1 of Technical Specification 3/4.4.9.3, and the associated portion of the Bases, being revised by changing the allowed time for charging pump swaps from ≤ 15 minutes to ≤ 1 hour.

4. The Bases associated with Technical Specification 3/4.4.1.3, page B 3/4 4-1, is changed to reflect change number 1 and to state that determination of the secondary side water temperature is required prior to starting the first reactor coolant pump.

5. The Bases associated with Technical Specification 3/4.4.9.3 is revised by:

   - adding a discussion of ASME Code Case N-640,

- adding a reference to WCAP-15570,
- adding a reference to Regulatory Guide 1.99, Revision 2,
- adding a discussion of boltup temperature,
- providing a clarification that deactivating all but one charging pump does not apply during charging pump swapping operations,
- adding a discussion of the PORV setpoint selection criteria, and
- changing K_{IR} to K_{IC} to reflect Code Case N-640.

There are also some editorial changes being made to the Bases. On page B 3/4 4-5, "FSAR" is changed to "UFSAR" for internal consistency. On page B 3/4 4-6, RT_{NTD} is changed to RT_{NDT} in three places to correct a typographical error.

6. Figure B 3/4 4-1, "Fast Neutron Fluence (E>1 Mev) As a Function of Full Power Service Life (EFPY)," is deleted since it is no longer used in Revision 2 of Regulatory Guide 1.99.

7. Figure B 3/4 4-2, "Effect of Fluence, Copper Content, and Phosphorus Content on ΔRT_{NDT} for Reactor Vessel Steels per Regulatory Guide 1.99," is replaced with a figure titled, "Predicted Decrease in Shelf Energy as a Function of Copper Content and Fluence." The figure replacement is being made because Regulatory Guide 1.99, Revision 2, uses the replacement figure to show the effect of fluence and copper content on upper shelf life for reactor vessel steel. The figure is also renumbered as B 3/4 4-1.

8. Table B 3/4.4-1, "Reactor Vessel Toughness Data (Unirradiated)," is updated to be consistent with the revised heatup and cooldown curves.

9. Figure B 3/4 4-3, "Isolated Loop Pressure-Temperature Limit Curve," is renumbered as B 3/4 4-2.

To meet format requirements the Index and Bases pages will be revised and repaginated as necessary to reflect the changes being proposed by this LAR.

B. DESIGN BASES

Overpressure protection for the RCS is achieved by means of self-actuated, steam safety valves located high in the system on the steam space of the pressurizer known as power operated relief valves (PORVs). These 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 overpressure protection enable temperature, the allowable system pressure is limited to 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 is at a temperature below the overpressure protection enable temperature. This creates the need for an overpressure protection system (OPPS) that 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.

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. This provides 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 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 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 violating the P/T limits does not compromise the integrity of the RCPB.

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 irradiation. 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 BVPS Unit No. 1 reactor pressure vessel, shows an increase in hardness and tensile properties and a decrease in ductility and toughness during high energy irradiation. The BVPS Unit No. 1 Vessel Radiation Surveillance Program, designed by Westinghouse, is described in WCAP-8457, "Duquesne Light Company Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program." 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 ensuring the integrity of reactor pressure vessels has been presented in Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code, "Fracture Toughness Criteria for Protection Against Failure." 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) which appears 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 by utilizing these allowable stress intensity factors. The use of K_{IC} adds substantial operating margin to the heatup and cooldown curves versus that provided by the stress intensity factor K_{IA}. Although K_{IA} was used for the current heatup and cooldown curves for BVPS Unit No. 1, the Bases references K_{IR}. In 1992 when Appendix G was moved from Section III to Section XI of the ASME Code, K_{IR} was changed to K_{IA}. Therefore K_{IR} is considered equivalent to K_{IA} throughout this license amendment request. Per ASME Code Case N-640, K_{IC} lower bound static fracture curve was allowed for determination of the heatup and cooldown curves in lieu of K_{IA} in ASME Section IX.

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 changes in mechanical properties of a given reactor pressure vessel steel, due to irradiation, can be monitored by a reactor vessel surveillance program, such as the BVPS Unit No. 1 Reactor Vessel Radiation Surveillance Program, in which a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. The increase in the average Charpy V-notch 30 ft-lb temperature, ΔRT_{NDT}, due to irradiation is added to the initial RT_{NDT}, along with a margin (M) to cover uncertainties, to adjust the RT_{NDT} for radiation embrittlement. Adjusted Reference Temperature (ART), defined as ART = initial RT_{NDT} + Margins for uncertainties + ΔRT_{NDT}, is used to index the material to the K_{IC} 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.

The design basis transients for determination of the PORV setpoint are the mass injection event and the heat addition scenario. These design basis transients are discussed in the following paragraphs. The PORV setpoint is selected in an attempt to meet generic guidelines such that:

1) the peak RCS pressure from the design basis mass injection and heat injection events will not exceed the maximum allowable pressure of the steady state 10 CFR 50 Appendix G heatup and cooldown reactor vessel limits or the discharge piping limit (800 psig), whichever is lower;

2) the minimum pressure from a design basis mass injection or heat injection event will not drop below the reactor coolant pump (RCP) No. 1 seal ΔP limit, and

3) dual (simultaneous) actuation of the OPPS PORVs is prevented.

The most severe credible mass input event utilized in the determination of the OPPS setpoint, involves single centrifugal charging pump operation. Specifically, a loss of air incident is postulated, whereby the flow control valve in the charging line fails open and, simultaneously, the flow control valve in the letdown line fails closed. Limiting mass input based on consideration of a single centrifugal charging pump operation is addressed in Note 1 of Technical Specification 3/4.4.9.3. The Note limits the amount of time two charging pumps can be capable of injecting into the RCS during low temperature conditions.

The heat addition scenario considered for analysis involves starting an idle RCP. The concern associated with starting an idle RCP is a heat input transient that could possibly overpressurize the RCS. This transient can occur when an idle RCP is started in a non-isolated loop whose secondary side water temperature is significantly higher than its primary side water temperature. However, adhering to a maximum secondary to primary side temperature difference and having an adequate pressure relief mechanism available; i.e., an operable PORV or an adequately sized RCS vent, will prevent overpressurization of the RCS. Technical Specification 3/4.4.1.3, "Reactor Coolant System – Shutdown," contains a note imposing a secondary to primary temperature difference restriction on starting an idle RCP. Adherence to this temperature difference restriction and the PORV and RCS vent requirements of Technical Specification 3/4.4.9.3 provides the necessary protection against the heat addition scenario.

C. JUSTIFICATION

The current heatup and cooldown curves are being revised to reflect Surveillance Capsule Y so that their applicability will change from 16 EFPY to 22 EFPY. The curves have been developed in accordance with the methodologies provided in WCAP-14040-NP-A, ASME Code Case N-640, and 1996 Addendum to the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G. The use of Code Case N-640, K_{IC} methodology, will result in a decrease in the likelihood of pump seal failure for operating plants. The primary reason for making this change 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 pump seal requirements is made larger and the chances of damaging the RCP seals and initiating a small break loss of coolant accident (LOCA), a potential pressurized thermal shock initiator, are reduced.

The OPPS design basis takes credit for the fact that overpressure events most likely occur during isothermal conditions in the RCS. As stated in WCAP 14040-NP-A, Revision 2, it is appropriate to utilize the steady-state Appendix G limit. 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 P/T 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 proposed enable temperature and the 50°F temperature difference allowance between the RCS and the steam generators prior to the start of an RCP is consistent with the OPPS analysis of record assumptions. In addition, OPPS provides for an operational consideration to maintain the integrity of the PORV piping. The setpoint is selected so that the 800 psig PORV piping limit is met.

During the development of this license amendment request, the approved Technical Specification Traveler Forms (TSTF) pertaining to Technical Specification 3/4.4.9.3 was reviewed for BVPS Unit No. 1 applicability. This review resulted in a portion of TSTF-285 being incorporated into Technical Specification 3/4.4.9.3 and the associated Bases.

Note 1 of Technical Specification 3.4.9.3 states that two charging pumps may be capable of injecting into the RCS for up to 15 minutes during pump swap operations. Incorporation of TSTF-285 changes the pump swap time to one hour. Fifteen minutes is insufficient time to prudently complete the operation of rendering the charging pump incapable of injection into the RCS. Closing and racking out valves, or racking out the pump breaker requires appropriate administrative controls to be followed by Operations personnel. With proper diligence, this action may or may not be accomplished in 15 minutes in all cases. One hour is a more reasonable allowable time period considering the small likelihood of an event during this brief period and the other administrative controls available (e.g., operator action to stop any pump that inadvertently starts). Therefore, it is acceptable to change the allowable pump swap time period to one hour.

D. SAFETY ANALYSIS

An analysis was performed to develop curves acceptable for heatup and cooldown based on the Surveillance Capsule Y Analysis. Subsequently, the PORV setpoint and OPPS enable temperature were generated applicable to the 22 EFPY heatup and cooldown curves that were developed as a result of the analysis of Surveillance Capsule Y.

Heatup and cooldown limit 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 ΔRT_{NDT} and adding margin.

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 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 steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99 Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (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 heatup and cooldown curves for normal operation.

Pressure and temperature (P/T) limit curves for normal heatup and cooldown of the RCS have been calculated using the methods outlined in WCAP-15570. These curves are developed using Code Case N-640 as noted in Reference 9. 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 of the ASME Code uses the reference stress intensity factor K_{IA} and is the basis for the current P/T limit curves. The proposed P/T limit 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 that is made when generating the revised P/T limits curve with K_{IC} is the lower bound fracture toughness curve selected. All other margins involved in the curve 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 P/T 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 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 P/T limit curves, and not K_{IA}.

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. This difference will be incorporated into plant procedure used to concur with the P/T limit requirements. The difference, however, has been incorporated into the generation of the OPPS setpoint. The heatup and cooldown curves also include the effect of the reactor vessel flange.

The heatup curve, Figure 3.4-2, was generated with no margins for instrumentation errors using heatup rates up to 100°F/hour applicable for the first 22 EFPY. Based upon the analysis documented in the BVPS Unit No. 1 reactor vessel stress report, the BVPS Unit No. 1 reactor vessel is structurally adequate for at least 200 cycles of plant heatups and cooldowns not exceeding 100°F/hour in accordance with the stress and fatigue analysis requirements of Class 1 components in Section III of the ASME Code. Therefore, actual heatup and cooldown rates of 100°F/hour are acceptable for BVPS Unit No. 1 from the standpoint of reactor vessel structural and fatigue analyses.

The cooldown curves, Figure 3.4-3, are also generated for first 22 EFPY with no margins for instrumentation errors for various cooldown rates up to 100°F/hr applicable for the first 22 EFPY. For both the heatup and cooldown curves, allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit curves. This is in addition to other criteria, which must be met before the reactor is made critical, as discussed in the following paragraph.

The reactor must not be made critical until P/T combinations are to the right of the critical limit curve of the heatup curve shown in Figure 3.4-2. The straight-line portion of the criticality limit is at the minimum permissible temperature (2485 psig) of the 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 Appendix G. 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 required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the

corresponding P/T curves for the heatup and cooldown calculated as described in WCAP-15570. For the heatup curve, the minimum temperature for the inservice hydrostatic leak tests for the BVPS Unit No. 1 reactor vessel at 22 EFPY is 289°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 curve defines all the above limits for ensuring prevention of nonductile failure for the BVPS Unit No. 1 reactor vessel during normal heatup and cooldown.

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

a) Two PORVs with a lift setting less than or equal to 432 psig , or

b) The RCS depressurized and an RCS vent of greater than or equal to 2.07 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 329°F, Mode 5, and Mode 6 when the reactor vessel head is on.

The PORV setpoint and OPPS enable temperature presently contained in Technical Specification 3.4.9.3 are being revised to be applicable to 22 EFPY and are based upon WCAP-14040-NP-A, ASME Code Case N-640, and 1996 Addendum to the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G. The current Maximum Allowable Nominal PORV setpoint was based on composite heatup and cooldown curves. The proposed PORV setpoint is based on isothermal conditions. The value of the PORVs setpoint is revised to 403 psig and the OPPS enable temperature is revised to 343°F. The PORV setpoint and OPPS enable temperature are applicable up to and including 30% steam generator tube plugging. The PORV setpoint conservatively accounts for instrument uncertainties associated with the wide range pressure transmitter (±70 psig), per Reference 6. The setpoint also accounts for the pressure difference between the wide-range pressure transmitter and the reactor vessel limiting beltline region, identified in Reference 8. The PORV setpoint has been selected such that the peak overpressure will not exceed the Appendix G limit or the 800 psig PORV piping limit without any restrictions placed on the number of RCPs in operation.

Thermal transport effects, which are applied to the heat injection transient results and account for the steam generator operation at a temperature 50°F higher than the reactor coolant temperature, have also been incorporated into the PORV setpoint. The heat addition event has been analyzed for RCS initial 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. A 100°F RCS temperature was used for the mass injection event. Pressure overshoots during the design basis events are based on a pressurizer PORV stroke open/close time of 3.0 seconds. The PORV setpoint is selected so that RCS pressures will not exceed the 22 EFPY Appendix G pressure limits down to the reactor vessel boltup temperature of 60°F.

During the PORV setpoint selection process it is noted that 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. This is due to the reset pressure being below the setpoint and 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 residual heat removal system. For conservatism, the second PORV is assumed to have failed. The maximum and minimum RCS pressures reached in the transient are a function of the selected setpoint and should fall within an acceptable pressure range.

The upper RCS pressure limit for OPPS is defined by Appendix G requirements or by PORV piping limitations, after consideration of all uncertainties and the ΔP between the wide range pressure transmitter and the reactor vessel limiting region. The lower limit on RCS pressure during the design basis OPPS mass injection and heat injection transients is established based on operational consideration for the RCP No. 1 seal limit which requires a nominal differential pressure 200 psid across the seal faces for proper film-riding performance. As part of the OPPS setpoint evaluation, margin to the RCP No. 1 seal limit is evaluated.

As demonstrated in Reference 6, pressure undershoot below the PORV setpoint during a design basis mass injection or heat injection event could cause the

differential pressure to be less than the minimum requirement of the RCP No. 1 seal. Therefore, there is the potential for RCS pressure to violate the RCP No. 1 seal limit at the lowest RCS temperatures. If this occurs, the method for choosing the PORV setpoint, which is consistent with approved methodology as defined in WCAP 14040-NP-A, Revision 2 is followed. Further, with the same setpoint for both PORVs, there is a potential for both valves to open simultaneously, then on closing, cause an undershoot that could violate the RCP No. 1 seal limit.

An envelope of mass injection rates was investigated to ensure that the worst case was considered for ultimate PORV setpoint determination. Based on this investigation PORV generic setpoint selection guidelines 2 and 3 could not be met. This is due to there being insufficient range between the upper Appendix G limit and the lower RCP seal limit to select PORV setpoints that provide protection against violating both limits. When this occurs, the setpoint selection that provides protection against violating the upper limit takes precedence as noted in WCAP 14040-NP-A.

The arming temperature (when the OPPS rendered operable) is established per ASME Section XI, Appendix G. At this temperature, a steam bubble would be present in the pressurizer, thus reducing the potential of a water hammer discharge that could challenge the piping limits. Based on this method, the arming temperature is > 343°F for 22 EFPY. The enable temperature is calculated based on either a RCS temperature of less than 200°F or materials concerns (reactor vessel metal temperature less than RT_{NDT} + 50°F), whichever is greater. The enable temperature does not address the piping limit issue. The calculated enable temperature is 308°F for 22 EFPY. As the arming temperature is higher and, therefore, more conservative than the enable temperature, the OPPS enable temperature is set to equal the arming temperature of 343°F for 22 EFPY.

Technical Specification 3.4.1.3, "Reactor Coolant System – Shutdown," is also changed to be consistent with the revised OPPS analysis. This Technical Specification contains a Note that imposes a RCP startup restriction based on a secondary to primary temperature difference. The current analysis assumes a 25°F difference in the temperature between the RCS and steam generators at the initiation of the heat injection event. The revised analysis considers a temperature difference of 50°F. The Note is, therefore, being revised to specify a temperature difference of 50°F.

The Bases associated with Technical Specification 3.4.1.3 is also revised to require the determination of the temperature of the steam generator prior to starting only the first RCP rather than each additional pump. The revised analysis demonstrates that reactor coolant system/steam generator fluid mixing occurs such that an equilibrium temperature condition is reached in less than five minutes. Therefore, it is not necessary to check secondary water temperatures before starting a second or third RCP after the initial pump start.

With respect to the charging pumps, a footnote is provided in Technical Specification 3/4.4.1.3 stating that two pumps are allowed to be capable of RCS injection for less than or equal to 15 minutes for pump swaps. This footnote is modified by changing the required time for charging pump swaps from ≤ 15 minutes to ≤ 1 hour. The 15 minute swap time is consistent with a time estimate made for the charging pump swap being performed by two qualified operations personnel. The time estimate starts with an open, racked out pump breaker on one pump and ends with an open, racked out, and properly surveilled pump breaker on the other pump. Experience has shown that 15 minutes to be overly burdensome and potentially adverse to safely and deliberately completing the pump swap actions. One hour is a more appropriate time period for these actions. It is noted that pump swapping actions are entered with the intent to minimize the actual time that more than one charging pump is physically capable of RCS injection.

E. NO SIGNIFICANT HAZARDS EVALUATION

The proposed license amendment is applicable to Beaver Valley Power Station (BVPS) Unit No. 1. It revises the reactor coolant system (RCS) pressure and temperature (P/T) limits to be valid through 22 effective full power years (EFPY). It also revises the overpressure protection system (OPPS) power operated relief valves (PORVs) setpoint and the OPPS enabling temperature. The proposed Unit No. 1 P/T limits incorporate the results from testing of Capsule Y described in WCAP-15571, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," Revision 0, November 2000. These changes have been prepared using the NRC-approved methodology described in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996, with two exceptions. They include the use of:

a) ASME Code Case N-640, "Alternate Reference Fracture Toughness for Development of P-T Curves for Section XI, Division 1," March 1999, and

b) ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," December 1995, (through 1996 Addendum).

Pursuant to 10 CFR 50.12(a)(2)(iii), FirstEnergy Nuclear Operating Company (FENOC) requests exemption to 10 CFR 50, Appendix G, based on American Society of Mechanical Engineers (ASME) Code Case N-640. The exemption request is included as Attachment C of license amendment request 292.

During the development of this license amendment request, the approved Technical Specification Traveler Forms (TSTF) pertaining to Technical Specification 3/4.4.9.3 was reviewed for BVPS Unit No. 1 applicability. This review resulted in a portion of TSTF-285 being incorporated into Technical Specification 3/4.4.9.3 and the associated Bases.

To meet format requirements the Index and Bases pages are also revised and repaginated as necessary to reflect the changes being proposed.

The no significant hazard considerations involved with the proposed amendment have been evaluated. The evaluation focused on the three standards set forth in 10 CFR 50.92(c), 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?

   No. The proposed changes do not result in physical changes being made to structures, systems, or components (SSCs), or to event initiators or precursors. Changing the heatup and cooldown curves, power operated relief valve (PORV) setpoint and overpressure protection system (OPPS) enable temperature to reflect 22 effective full power years (EFPY) will not affect the ability of the OPPS to control the reactor coolant system (RCS) at low temperatures such 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.

   Also, the proposed changes do not impact the design of plant systems such that previously analyzed SSCs would now be more likely to fail. The initiating conditions and assumptions for accidents described in the Updated Final Safety Analysis Report (UFSAR) remain as previously analyzed. Thus, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

   The proposed changes do not alter any assumptions previously made in the radiological consequence evaluations nor affect mitigation of the radiological consequences of an accident described in the UFSAR. As such, the consequences of accidents previously evaluated in the UFSAR will not be increased and no additional radiological source terms are generated. Therefore, there will be no reduction in the capability of those SSCs in limiting the radiological consequences of previously evaluated accidents and reasonable assurance that there is no undue risk to the health and safety of the public will continue to be provided. Thus, the proposed changes do not involve a significant increase in the consequences of an accident previously evaluated.

   Therefore, the proposed changes do not involve a significant increase in 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?

   No. The proposed changes do not involve physical changes to analyzed SSCs or changes to the modes of plant operation defined in the technical specification. The proposed changes do not involve the addition or modification of plant equipment (no new or different type of equipment will be installed) nor do they alter the design or operation of any plant systems. No new accident scenarios, accident or transient initiators or precursors, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes.

   The proposed changes do not cause the malfunction of safety-related equipment assumed to be operable in accident analyses. No new or different mode of failure has been created and no new or different equipment performance requirements are imposed for accident mitigation. As such, the proposed changes have no effect on previously evaluated accidents.

   Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

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

   No. The proposed changes have been determined through supporting analyses to be in accordance with the methodologies set forth in the regulations. Compliance with NRC approved methodologies provide for an adequate margin of safety and 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 as described in the UFSAR.

   The new heatup and cooldown curves define the limits for ensuring prevention of nonductile failure for the BVPS Unit No. 1 reactor vessel and do not significantly reduce the margin of safety for the plant.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

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.

References

1. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 2, January 1996.

2. ASME Code Case N-640, "Alternate Reference Fracture Toughness for Development of P-T Curves for Section XI, Division 1," March 1999.

3. ASME Boiler and Pressure Vessel Code, Section XI, "Rule for Inservice Inspection of Nuclear Power Plant Components," Appendix G, "Fracture Toughness Criteria for Protection Against Failure," December 1995, (through 1996 Addendum).

4. WCAP-15571, "Analysis of Capsule Y from Beaver Valley Unit 1 Reactor Vessel Radiation Surveillance Program," Revision 0, November 2000.

5. WCAP-15570, "Beaver Valley Unit 1 Heatup and Cooldown Limit Curves for Normal Operation," Revision 2, April 2001.

6. "Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company – Overpressure Protection System – Setpoints for Y-Capsule," Revision 1, April 2001.

7. Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2.

8. Westinghouse Nuclear Safety Advisory letter NSAL-93-005A, "Cold Overpressure Mitigation System (COMS) Nonconservatism," dated March 10, 1993.

9. EDRE-SMT-98-135, "Technical Basis for Revised P-T Limit Curve Methodology," W. H. Bamford et al, December 4, 1998.

ATTACHMENT C

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 292
10 CFR 50, Appendix G Exemption Request

Requirement for Which Exemption is Requested

Pursuant to 10 CFR 50.12, "Specific Exemptions," enclosed is a request for exemption from certain requirements of 10 CFR 50.60, "Acceptance Criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," and 10 CFR 50, Appendix G, "Fracture Toughness Requirements." This exemption is requested to allow the application of ASME Code Case N-640 in determining the acceptable overpressure protection system (OPPS) power operated relief valve (PORV) setpoints.

ASME Section XI Code Requirements

ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure," Article A-2000, provides the service limits for pressure vessels, and establishes the allowable vessel loading (internal pressure, external load, thermal stress) versus temperature. The code requirements are to maintain vessel operation conditions within Article A-2000 requirements.

Code Requirement from Which Exemption is Requested

Exemption is requested from 10 CFR 50, Appendix G, and ASME Section XI, Appendix G, requirements for reactor vessel pressure limits at low temperature.

Basis for Exemption Request

Current OPPS setpoints produce operational constraints by limiting the pressure and temperature (P/T) range available to the operator to heat up and cool down the plant. The "operating window" through which the operator must heat up and pressurize, or cool down and depressurize the reactor coolant system (RCS) is determined by Appendix G of Section XI, and the minimum required pressure for the reactor coolant pump (RCP) No. 1 seal, adjusted for OPPS overshoot and instrument uncertainties. Under the P/T requirements in Appendix G of ASME Section XI, OPPS can have significant impact on operation by limiting RCP operation at low temperatures. In addition, the operating pressure window imposed by OPPS becomes more and more restrictive with reactor vessel service. Reducing this operating window could potentially have an adverse safety impact by increasing the possibility of inadvertent OPPS actuation due to pressure surges associated with normal plant evolutions such as

RCS start and swapping operating charging pumps with the RCS in a water-solid condition.

The impact on the P/T limits and OPPS setpoints has been evaluated due to increasing the service period to 22 Effective Full Power Years (EFPY) based on ASME Section XI, Appendix G, requirements. The results indicate OPPS would significantly restrict the ability to perform plant heatup and cooldown, create an unnecessary burden to plant operations, and challenge control of plant evolutions required with OPPS enabled.

Proposed Alternative

The use of ASME Code Case N-640 requirements for reactor vessel pressure limits at low temperatures is proposed as an alternative to 10 CFR 50, Appendix G, and ASME Section XI, Appendix G, requirements.

Justification for Granting Relief

The special circumstances associated with this exemption request in accordance with 10 CFR 50.12(a)(2)(iii) are associated with use of the current K_{IR}, instead of K_{IC} allowed by ASME Code Case N-640. Using K_{IR} would make the P/T Limit Curves overly conservative, since the K_{IR} stress intensity is based on both static and dynamic fracture toughness data, while the K_{IC} stress intensity is based on only static fracture toughness data.

The underlying purpose of 10 CFR 50.60, Appendix G, is to establish fracture toughness requirements for ferritic material of pressure retaining components of the reactor coolant pressure boundary. These requirements provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences, to which the pressure boundary may be subjected to over its service lifetime. Section IV.A.2 of this appendix requires that the reactor vessel be operated with P/T limits at least as conservative as those obtained by following the methods of analysis and the required margins of safety of Appendix G of ASME Section XI.

Appendix G of Section XI of the ASME Code requires the P/T limits are calculated:

a) using a safety factor of 2 on the principal membrane (pressure) stresses,

b) margin added to the reactor vessel RT_{NDT} in accordance with Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials,"

c) assuming a flaw at the surface with a depth of 1/4 of the vessel wall thickness and a length of 6 times its depth, and

d) using conservative fracture toughness curves that are based on the lower bound of static, dynamic, and crack arrest fracture toughness tests on material similar to the reactor vessel material.

The proposed amendment to the P/T Limit Curves relies in part on this requested exemption. These revised P/T Limit Curves, as specified in ASME Code Case N-640, use a lower stress intensity factor, K_{IC} instead K_{IR}, which results in higher allowable pressures. K_{IR} is a reference stress intensity factor based on the lower band values of K_{IC} and K_{IA}.

10 CFR 50 Appendix G requires the use of the K_{IR} stress intensity, which is based on both static and dynamic fracture toughness data. ASME Code Case N-640 allows the use of the K_{IC} stress intensity, which is based on only static fracture toughness data.

With the exception of the application of ASME Code Case N-640, all other requirements of 10 CFR 50 Appendix G are satisfied.

Use of the K_{IC} in determining the lower bound fracture toughness in the development of P/T Limit Curves is more technically correct than the K_{IA} curve. Note that K_{IA} was used for the current heatup and cooldown curves for BVPS Unit No. 1. In 1992 when Appendix G was moved from Section III to Section XI of the ASME Code, K_{IR} was changed to K_{IA}. The K_{IC} curve appropriately implements the use of static initiation fracture toughness behavior to evaluate the integrity of the reactor vessel.

Furthermore, there are no known mechanisms of degradation for this region, other than fatigue. The calculated design fatigue usage for this region is less than 0.1, so it may be concluded that flaws are unlikely to initiate in this region.

Based on the aforementioned information, FENOC requests an exemption to 10 CFR 50.60(a), based on American Society of Mechanical Engineers (ASME) Code Case N-640.

ATTACHMENT D

Beaver Valley Power Station, Unit No. 1
License Amendment Request No. 292

Proprietary (Class 2C) and Non-proprietary (Class 3) Version of Westinghouse Report
"Beaver Valley Unit 1 FirstEnergy Nuclear Operating Company --
Overpressure Protection System -- Setpoints for Y-Capsule"
Revision 1, April 2001