ComEd
An Exelon Company

Commonwealth Edison Company
1400 Opus Place
Downers Grove, IL 60515-5701
www.exeloncorp.com

RS-00-132

November 27, 2000

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Response to Request for Additional Information Regarding the License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

References: (1) Letter from R. M. Krich (Commonwealth Edison Company) to U.S. NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," dated July 5, 2000

(2) Letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron and Braidwood - Request for Additional Information Regarding the Power Uprate Request," dated October 19, 2000

In Reference 1, we submitted proposed changes to Facility Operating License Nos. NPF-72, NPF-77, NPF-37 and NPF-66, and Appendix A, Technical Specifications (TS), for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively. The proposed changes would revise the maximum power level specified in each unit's license and the TS definition of rated thermal power.

On September 20, 2000, members of Commonwealth Edison (ComEd) Company and the NRC met to discuss technical issues associated with this license amendment request. In Reference 2, the NRC requested that we formally document the information discussed during this meeting along with some additional information in order to complete their evaluation. In a subsequent teleconference on November 8, 2000, the NRC also requested that additional information be provided regarding the documents reviewed in support of the Power Uprate Environmental Assessment. Attachment 1 to this letter provides our response to these requests for additional information.

A001

The NRC requested that this response be submitted by November 20, 2000. In a telephone conversation between K. A. Ainger (ComEd) and A. J. Mendiola (NRC), the requested date for this response was extended to November 27, 2000.

Should you have any questions or concerns regarding this information, please contact Mr. J. A. Bauer at (630) 663-7287.

Respectfully,

K. A. Ainger for

R. M. Krich
Director – Licensing

Attachment 1: Response to Request for Additional Information Regarding a License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Braidwood Station
NRC Senior Resident Inspector – Byron Station
Office of Nuclear Safety – IDNS

STATE OF ILLINOIS	)	
COUNTY OF DUPAGE	)	
IN THE MATTER OF	)	
COMMONWEALTH EDISON (COMED) COMPANY	)	Docket Numbers
BYRON STATION UNITS 1 AND 2	)	STN 50-454 AND STN 50-455
BRAIDWOOD STATION UNITS 1 AND 2	)	STN 50-456 AND STN 50-457

SUBJECT: **Response to a Request for Additional Information Regarding a Previous ComEd License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations**

AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Director – Licensing

Subscribed and sworn to before me, a Notary Public in and for the State above named, this 22 day of November, 2000.

Notary Public

OFFICIAL SEAL
JEFFERY A BATARA
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES:02/24/02

ATTACHMENT 1

Response to Request for Additional Information Regarding a License Amendment Request to Permit Uprated Power Operations at Byron and Braidwood Stations

In a letter from G. F. Dick (U.S. NRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron and Braidwood - Request for Additional Information Regarding the Power Uprate Request," dated October 19, 2000, the NRC requested that the below additional information be provided.

NRC Question Set A

A.1 *It does not appear that there is any discussion addressing how the power uprate at Byron and Braidwood impacts the existing analysis performed for station blackout. Please discuss and verify the assumptions for the existing station blackout analysis are valid for the power uprate conditions, particularly as they relate to issues such as heat-up analysis, equipment operability and battery capacity.*

A.1 Response

Evaluations of the systems impacted by the uprate did not identify any changes to assumptions, design, or operating conditions that adversely affect the ability to provide safe shutdown for a Station Blackout (SBO). In addition, power uprate does not create any additional electrical demands or equipment modifications that would impact any plant heatup analysis or increase battery loading. Power uprate will result in increased decay heat load during the coping period.

The increased decay heat will require an increase in the total volume of water that would be supplied by the Auxiliary Feedwater (AFW) System, during the coping period. However, sufficient useable inventory in the Condensate Storage Tank (CST) is available to satisfy AFW requirements for plant cool-down.

Following is a review of the SBO coping assessment:

1. Condensate Inventory for Decay Heat Removal

   The analysis performed for the AFW storage (i.e., License Report Section 4.2.3.4) was performed for the limiting transient with respect to the CST decay heat removal requirements during the coping period. This analysis is based on the loss-of-offsite power (LOOP) transient and the inventory requirement to bring the unit from full power to a hot standby condition, maintain the plant at hot standby for four hours, and then cool the reactor coolant system (RCS) to the residual heat removal (RHR) cut-in temperature (i.e., 350°F) in four hours. The result is a required useable inventory less than the existing Technical Specification of 200,000 gallons. This requirement also bounds the SBO required inventory for decay heat removal without cooldown for four hours.

2. Effects of Loss of Ventilation

   The increase in power does not impact the loss of ventilation or result in increased heat loads to areas serving SBO components, therefore no additional analysis was required as a result of the power uprate. (Refer to License Report Section 9.3.18).

3. Reactor Coolant Inventory

   The alternate alternating current (AC) power source to the required reactor coolant make-up systems is not impacted by an increase in power; therefore, no additional analyses were required as a result of the power uprate.

4. Containment Isolation

   There are no changes to containment isolation requirements or capabilities as a result of the increase in power level. Therefore no additional analyses were required as a result of the power uprate.

5. Alternate AC Power Source SBO Loading

   No new or increased Emergency Diesel (EDG) electrical loads are required as a result of the increase in power level. The operation or capacities of components associated with SBO are not impacted by increase in power. The diesel driven auxiliary feedwater pump can supply condensate for decay heat removal. Therefore no additional analysis was required as a result of the power uprate. The Alternate AC power source will continue to provide normal charging to one of Engineered Safety Features (ESF) batteries per unit upon crosstieing the non-blackout unit Alternate AC EDG to the blackout unit.

6. Analysis of Essential Service Water (i.e., SX) Crosstie between Units

   Since the increase in decay heat load can be accommodated by the CST inventory during the four hour coping duration, the load on the SX system is not impacted. Therefore no additional analysis was required as a result of the power uprate.

A.2 *It is indicated in Page 9-126 of the power uprate licensing report that the electrical equipment located inside and outside containment, that performs a safety-function, must remain qualified for the accident, pressure, and humidity environment at the uprate power conditions. However, the impact of power uprate on humidity environment during normal and accident conditions were not provided. Provide the impact of power uprate on humidity environment for electrical equipment inside and outside containment.*

A.2 Response

Table 3.11-2 of the Byron and Braidwood Stations Updated Final Safety Analysis Report (UFSAR), "Plant Environmental Conditions," provides the normal and

accident humidity conditions. Power Uprate does not affect the normal humidity values for the plant Environmental Qualification (EQ) Zones, presented in this table. Those EQ Zones subject to accident conditions are likewise not affected by power uprate, as the existing analysis assumes the maximum accident relative humidity in these areas is 100%. Thus the original environmental qualification basis with respect to humidity remains unaffected.

For additional information on the review of uprate on heating, ventilation, and air conditioning refer to Sections 9.3.18 and 9.3.12 (i.e., Containment Cooling) of the Licensing Report.

A.3 *Explain, in detail, the bounding conditions of the containment revised temperature/pressure profiles as shown in Figures 9.3.21-1 and 9.3.21-2.*

<u>A.3 Response</u>

The bounding conditions presented in Licensing Report Figures 9.3.21-1 and 9.3.21-2 were developed based on the revised analyses performed in support of the Byron and Braidwood Stations Steam Generator Replacement Project. These analyses remain bounding for uprated power conditions.

These bounding conditions are again presented below as Figures A.3-1 and A.3-2. Specific points on these bounding curves are noted below.

120°F to 321°F (0 to 27 seconds),
321°F to 333°F (27 to 61 seconds),
333°F (61 to 65 seconds),
333°F to 320°F (65 to 83 seconds),
320°F (83 to 180 seconds),
320°F to 270°F (3 to 5 minutes),
270°F (5 to 20 minutes),
270°F to 170°F (20 minutes to 1 day),
170°F to 155°F (1 to 20 days), and
155°F (20 days to 1 year).

The Environmental Qualification (EQ) profiles for Unit 1 of Byron and Braidwood Stations, and Unit 2 of Byron and Braidwood Stations are presented in Figures A.3-1 and A.3-2, respectively. The following curves are provided for each unit as noted in the legend of each figure.

- Original EQ Curve: This curve presents the temperature vs. time profile for the EQ envelope developed as part of the original EQ Design Basis.
- Bounding EQ Curve: This curve represents the temperature vs. time profile for the revised EQ envelope, as described above, was developed as part of the reanalyses for the Steam Generator Replacement Project, December 29, 1999.

- Original Steam Generator (OSG) Composite: This curve represents the temperature vs. time profile for a double ended break of a steam line inside containment for the original steam generators.

- Split Rupture: This partial curve represents the temperature vs. time profile for the uprate that is outside the OSG composite.

- Double Ended Hot Leg (DEHL) Minimum (Min) Safety Injection (SI): This curve represents the uprate temperature vs. time profile for a double ended break of the reactor coolant piping between the reactor and a steam generator with a single train of ECCS available.

- Double Ended Pipe Suction (DEPS) Min SI: This curve represents the uprate temperature vs. time profile for profile for a double ended break of the reactor coolant piping between a steam generator and a reactor coolant pump with a single train of ECCS available.

- DEPS Maximum (Max) SI: This curve represents the uprate temperature vs. time profile for profile for a double ended break of the reactor coolant piping between a steam generator and a reactor coolant pump with both trains of Emergency Core Cooling System (ECCS) available.

Figure A.3-1
EQ Profile Unit 1

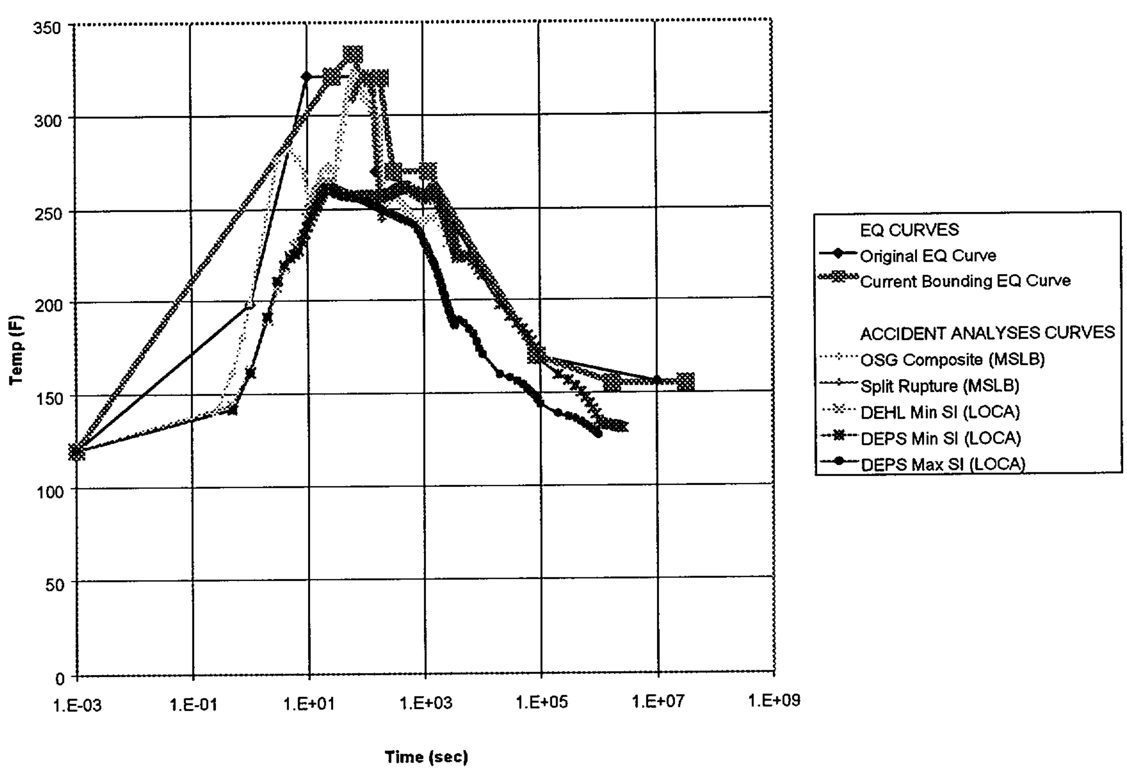
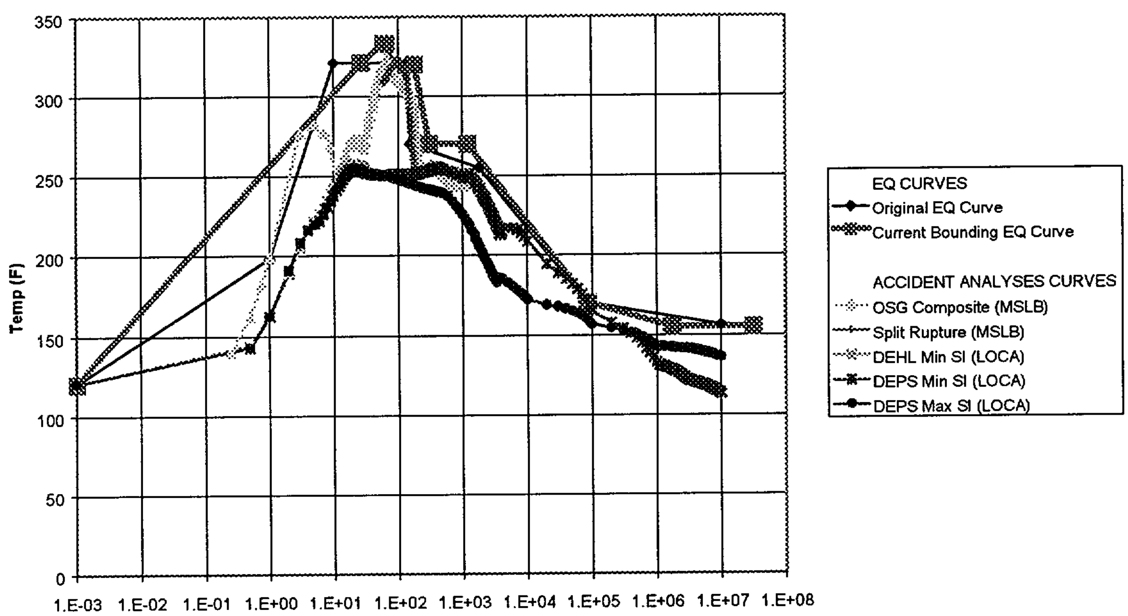


Figure A.3-2
Unit 2 EQ Profile



A.4 *For electrical equipment within the scope of 10 CFR 50.49 outside the containment, explain, in detail, how the electrical equipment remains qualified when the peak compartment temperature of 518.4 degrees Fahrenheit (°F) from Case 102-L (Byron/Braidwood, Unit1) and the peak compartment temperature of 502.5 °F from Case 102-M (Byron/Braidwood, Unit 2) exceed the temperature of 419 °F previously used to demonstrate equipment qualification. In addition, explain how the post-accident monitoring equipment remains qualified when the peak long-term temperature used for evaluation of post-accident monitoring equipment outside containment in the steam tunnels and valve room exceeds the current peak temperature of 515.25 °F by 3 °F. In addition, please provide the temperature for which the post-accident monitoring equipment was previously qualified.*

<u>A.4 Response</u>

Tables 6.5.5-4 through 6.5.5-7 of the Licensing Report show the compartment peak temperature results for Unit 1 at Byron and Braidwood Stations. Case 70-C with a Main Steamline Isolation Valve (MSIV) failure (i.e., 0.3 ft^2 break case) yields a peak temperature of 396.0°F, at the time of steamline isolation. The overall peak compartment temperature is 518.4°F from Case 102-L with an MSIV failure (i.e., 1.2 ft^2 break case).

Tables 6.5.5-8 through 6.5.5-11 of the Licensing Report show the compartment peak temperature results for Unit 2 at Byron and Braidwood Stations. Case 70-D with an MSIV failure (i.e, 0.4 ft^2 break case) yields a peak temperature of

413.5°F, at the time of steamline isolation. The overall peak compartment temperature is 502.5°F for Case 102-M with an MSIV failure (i.e., 1.4 ft^2 break case).

Note per the Braidwood Station Safety Evaluation (i.e., BRW-SE-1997-201/6G-97-0105) supporting UFSAR Design Review Package (DRP) 7-040 states, "Qualification of the safe shutdown equipment is not a concern since the revised safe shutdown temperature is less than the previous temperature of 419°F used to demonstrate qualification." "....The revised analysis also determined that the peak temperature was reduced to 515.25°F (i.e., 515°F). Although lower than previous determined, the qualification acceptability of the Regulatory Guide 1.97 equipment affected by the Main Steamline Break (MSLB) revised peak temperature is summarized in the attached evaluation."

Note that equipment required for safe shutdown (e.g., Main steamline isolation) is qualified for the peak temperature reached at the time of steamline isolation. The post accident monitoring equipment is qualified for the long term peak temperature. The uprate results increased the peak temperature at the time of steamline isolation to 414°F (i.e., 413.5°F), but remains less that the 419°F previously evaluated. The peak long term compartment temperature increased to 518°F (i.e., 517.72°F). Calculation BRW-96-550-E, "Evaluation of the Environmental Effects of a Main Steam Line Break Outside Containment," performed at 517.72°F would also apply to 518°F with no impact on environmental qualification due to the small temperature difference, margin/conservatism in the analysis and most importantly the significant margin which exists between the test vs. postulated plant conditions.

NRC Question Set B

In its submittal the licensee has stated, "Reactor trip and Engineered Safety feature (ESF) actuation setpoints have been assessed and no needed changes were identified as a result of uprated power operations.... All acceptance criteria including those for LBLOCA [large break loss-of-coolant accident], SBLOCA [small break loss-of-coolant accident], non-LOCA [loss-of-coolant accident] accidents, containment pressure and temperature, and radiological dose limits, continue to be met.... The results of the NSSS [nuclear steam supply system] analysis demonstrated that all systems were capable of performing their current design basis functions with either no changes or with appropriate changes to programs, setpoints and alarms." (Reference Attachment A to the submittal Pages A-2 and A-15.)

In regard to the above statements, please furnish the following additional information:

B.1 *Identify safety-related instrumentation and control (I&C) functional units for which the power uprate evaluation indicated no change was needed in trip setpoint and allowable values for uprated power operation.*

B.1 Response

The following tables list the I&C functional units for which no changes were needed due to power uprate. The Technical Specification allowable values are provided for information.

The trip setpoints for the unaffected safety-related I&C functions for uprated power operation can be found in the existing Byron Station and Braidwood Station Technical Requirements Manuals, Sections 2.0.a and 2.0.b (i.e., Tables T2.0.a-1 and T2.0.b-1).

From Technical Specification Table 3.3.1-1
Reactor Trip System Instrumentation

Function	Allowable Value	Change
Manual Reactor Trip	NA	
Power Range Neutron Flux High Low	 ≤110.8% RTP (High) ≤27.0% RTP (Low)	No Change
Pressurizer Pressure	≥ 1875 psig (Low) ≤ 2393 psig (High)	No Change
Pressurizer Water Level	≤ 93.5% instrument span (High)	No Change
Overtemperature ΔT Overpower ΔT	As found in the COLR	No Change
Reactor Coolant Flow	≥ 89.3% of loop minimum measured flow (Low)	No Change
Power Range Neutron Flux Rate High Positive High Negative	 ≤6.2% RTP with TC ≥ 2 sec ≤6.2% RTP with TC ≥ 2 sec	No Change
Intermediate Range Neutron Flux	≤30.0% RTP	No Change
Source Range Neutron Flux	≤ 1.42 E5 cps	No Change
Undervoltage RCPs	≥ 4920 v	No Change
Underfrequency RCPs	≥ 56.08 Hz	No Change
SG Water Level – Low Low Unit 1 Unit 2	 ≥ 16.1% of NR Instr. Span ≥ 34.8% of NR Instr. Span	No Change
Turbine Trip Emergency Trip Hdr Throttle valve closure	 ≥ 910 psig ≥ 1% Open	No Change
Safety Injection Input from ESFAS	NA	
Reactor Trip System Interlocks Low Power Rx Trips Block P-7 P-10 Input P-13 Input SR Block Permissive P-6 Power Range Neutron Flux P-8 Power Range Neutron Flux P-10 Turbine Impulse Pressure P-13	 NA NA ≥ 6E-11 amp ≤ 32.1%RTP ≥ 7.9% and ≤ 12.1% RTP ≤ 12.1% turbine power	No Change

Reactor Trip Breakers	NA	No Change
Reactor Trip Breaker UV and Shunt Trip	NA	No Change
Automatic Trip Logic	NA	No Change

From Technical Specification Table 3.3.2-1
Engineered Safety Feature Actuation System Instrumentation

Function	Allowable Value	Change
Safety Injection Manual Initiation Automatic Actuation Logic Containment Pressure High Pressurizer Pressure Low Steam Line Pressure Low	 NA NA ≤ 4.6 psig ≥ 1817 psig ≥ 614 psig (time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds)	No Change in allowable value. Safety analysis limit for steam line pressure low is revised as follows: MSLB inside containment – no change. MSLB outside containment – revised to 503 psia. MSLB core response – revised to 450 psia. FLB – no change.
Containment Spray Manual Initiation Automatic Actuation Logic Containment Pressure High-3	 NA NA ≤ 21.2 psig	No Change in allowable value. Safety analysis limit for containment pressure high is revised from 24.8 psig to 24.0 psig.
Containment Isolation Phase A Isolation Manual Automatic Phase B Isolation Manual Automatic Containment Press. High-3	 NA NA NA NA ≤ 21.2 psig	No Change
Steam Line Isolation Manual Automatic Actuation Containment Pressure High-2 Steam Line Pressure Low Negative Rate High	 NA NA ≤ 9.4 ≥ 614 psig (time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds) ≤ 165.3 psi (time constants used in the rate/lag controller are $t_1 \geq 50$ seconds)	No Change in allowable value. Safety analysis limit for steam line pressure low is revised as follows: MSLB inside containment – no change. MSLB outside containment – revised to 503 psia. MSLB core response – revised to 450 psia. FLB – no change.

Turbine Trip and Feedwater Isolation Automatic Actuation Logic SG Water Level High High P-14 Unit 1 Unit 2 Safety Injection	 NA ≤ 89.9% NR Span ≤ 81.5 % NR Span Refer to SI function	No Change
Auxiliary Feedwater Automatic Actuation Logic SG Water Level Low Low Unit 1 Unit 2 Safety Injection Loss of Offsite Power UV Reactor Coolant Pump Aux Feedwater Pp Suction Xfer	 NA ≥ 16.1% NR Span ≥ 34.8 % NR Span Refer to SI Function ≥2730 V ≥ 4920 V ≥ 17.4 psia	No Change
Switchover to Containment Sump Automatic Actuation Logic RWST Level Coincident with Safety Injection	 NA ≥ 44.7% of Span Refer to SI Function	No Change
ESFAS Interlocks Reactor Trip P-4 Pressurizer Pressure P-11 Tavg Low Low P-12	 NA ≤ 1936 psig ≥ 548.0 °F	No Change.

B.2 *Identify-safety related I&C functional units where appropriate changes to programs, setpoints and alarms are being made and provide a short description of changes.*

<u>B.2 Response</u>

Scaling and setpoint changes that are being performed at the stations to support the uprated conditions include the below items. Note that items typically performed at the station, e.g., precision calorimetrics, nuclear instrumentation adjustments, etc., are not reflected on the list.

ATTACHMENT 1
(continued)

Table B.2
Scaling and Setpoint Changes

I&C Change	Description of Change
HP Turbine 1st Stage Pressure	The High Pressure Turbine Impulse Pressure Transmitters will be rescaled to reflect the change in pressure due to uprated power conditions. Where necessary, impulse pressure transmitters will be replaced to support scaling to the revised span.
Steam Generator Narrow Range Level	Due to RCS coolant temperature program changes, steam pressure will change resulting in a density change to the SG shell side water. Narrow range SG level transmitters will be rescaled to reflect the change in water density so accurate water level indication is maintained.
Reactor Cooling System Loop Temperatures	Reactor Coolant Delta T and Tave instrumentation will be rescaled to allow operating at uprated power conditions. Delta T indication will be re-normalized to uprated calorimetric power indication and programmed RCS temperature control (Tref) will be adjusted to operate at a new full power Tave. The OTΔT f1(ΔI) penalty and penalty bands will be scaled into the Delta T protection instrumentation. The following control systems are also adjusted any time a change in RCS operating temperature is implemented at Byron/Braidwood: Pressurizer Level Program to reflect change in water expansion from zero power to full power, Hi Tave Alarm, and Steam Dump control system
Low Temperature Operating Pressure Setpoints (LTOPS)	The Low Temperature Overpressure Protection System setpoints will be revised based on Westinghouse evaluation of reactor vessel fluence. The revised setpoints will be documented in a revised Pressure/Temperature Limits Report (PTLR).
Pressure Relief Tank (PRT) Level Alarms	The PRT high and low level alarms are being adjusted due to revised setpoint analysis performed by Westinghouse. The revised setpoints provide more operating margin for the PRT
Steam Pressure Header	The unit 2 low steam pressure alarm will be adjusted for the lower steam pressure.
Unit 2 Steam Generator FW High Flow Alarm	The Unit 2 S/G main nozzle high flow alarm will be adjusted to reflect the increased feedwater flow due to uprated power conditions.
Radiation Monitor Setpoints	Adjustment of several radiation monitor setpoints to support the power uprate analysis performed.

Note: Where the setpoint change impacts an annunciator or computer point/parameter, the associated point/parameter will also be revised.

B.3 *Provide a brief description of setpoint calculation methodology used for assessing acceptability of trip actuation setpoints and allowable values of safety-related instruments for uprated power operation, and confirm that uncertainty values were established with a 95 percent probability at a 95 percent confidence level. Also provide reference(s), if the setpoint methodology used at Byron and Braidwood has been reviewed and approved by the staff.*

<u>B.3 Response</u>

The determination of instrument setpoints is based on plant operating experience, conservative licensing analysis, and/or limiting design/operating values. The original setpoint methodology used by Commonwealth Edison (ComEd) Company for Byron Station and Braidwood Station that established the original Reactor Protection System (RPS)/Engineered Safety Features Actuation System (ESFAS) trip setpoints was the Westinghouse methodology (WCAP-12523 – square root of the sum of the squares method, (i.e, SRSS)) that was approved by the NRC. This does differ from the method discussed in American National Standards Institute/Instrument Society of America (ANSI/ISA) S67.04, Parts 1 and 2, which did not exist at the times the Byron Station and Braidwood Station licenses were issued; however, the original Westinghouse methodology is still used for RPS/ESFAS setpoints. Note that no RPS/ESFAS setpoints have been changed as a result of power uprate.

Instrument uncertainties and allowable values for BOP setpoints are established using the ComEd setpoint methodology as outlined in our corporate procedure Nuclear Engineering Standard (NES) - Electrical Instrumentation and Control (EIC)-20.04, "Analysis of Instrument Channel Setpoint Error and Instrument Loop Accuracy." Each setpoint is selected with sufficient margin between the nominal system setting and the safety analysis value (i.e., analytical limit). This precludes inadvertent initiation of the protective functions, while assuring adequate allowances for inherent instrument inaccuracies, calibration uncertainties, drift, process errors, and applicable design basis events (i.e., normal and accident), relative to the analytical limit.

The ComEd setpoint methodology is based on ANSI/ISA S67.04, Parts 1 and 2 of the 1994 version. This methodology does not deviate from nor require ComEd to take any exceptions from Regulatory Guide 1.105, Revision 1, except for those noted in the Byron and Braidwood Updated Final Safety Analysis (UFSAR)[1]. The ComEd methodology establishes uncertainty values at a 95 percent probability and a 95 percent confidence level.

The NRC inspected the ComEd setpoint methodology as part of an instrumentation and control inspection at Dresden Nuclear Power Station in 1994. An evaluation of this methodology was also included in Amendment No. 129 to Facility Operating License No.NFP-11 for LaSalle County Station. In this amendment, "the staff compared the methodology used in Calculation No. L-001420 to the methodology shown in licensee document NES-EIC-20.04. The staff determined that the methodology as shown in NES-EIC-20.04 and Appendix A, B and C of that document was suitable for use in Calculation No. L-001420 because it contained the proper terms for establishing setpoints.

As a result of the increase in the reactor power, all potentially affected analytical limits for setpoints were assessed. Analytical limits potentially affected by the power uprate are identified below. Actual plant setpoint adjustments will provide adequate allowances between the operational settings and the analytical limits to ensure the necessary safety functions.

Whenever setpoints were adjusted, the existing calculation (or affected portion) was modified with the new design input. Thus no new methodologies were introduced. Any new setpoint or scaling calculations done during the implementation phase of power uprate will be based on the existing methodology.

Note 1: Position C.5, Locking devices on instrument setpoint adjustment mechanisms. Position C.6, requires documentation of the assumptions used in selecting setpoint values and the margins between the setpoints and the limiting safety system values. The documentation is to include definition of instrument setpoint drift rate and the relationship of the drift rate to testing intervals. The Byron/Braidwood design conforms to this position only to the degree that setpoints are documented on the instrument data sheets along with instrument range and the maximum range of the parameter being measured. With respect to the other requirements of Position C.6, generic drift rates are not generally available for any instruments since drift rates would be affected by the particular service to which the instrument was subjected.

<u>NRC Question Set C</u>

C.1 *The licensee stated that certain WCAP reports provide the details of how the increase in core rated power will effect the pressure-temperature (P-T) limits and pressurized thermal shock (PTS) evaluations for the Byron and Braidwood units; however, it is not clear that the reports have been placed on the dockets for Byron and Braidwood. If these WCAPs have not been placed on the dockets for Byron and Braidwood, please provide the following documents: WCAP-15391 and WCAP-15391 on P-T limit curves and WCAP-15389 and WCAP-15390 on PTS evaluations for the Byron units; WCAP-15364 and WCAP-15373 on P-T limit curves and WCAP-15365 and WCAP-15381 on PTS evaluations for the Braidwood units.*

<u>C.1 Response</u>

The following WCAPs are provided for information in Attachment C.1 at the end of this document:

- WCAP-15391, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation"
- WCAP-15392, "Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation"
- WCAP-15390, "Evaluation of Pressurized Thermal Shock for Byron Unit 1"
- WCAP-15389, "Evaluation of Pressurized Thermal Shock for Byron Unit 2"
- WCAP-15364, "Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation"

- WCAP-15373, "Braidwood Unit 2 Heatup and Cooldown Limit Curves for Normal Operation"
- WCAP-15365, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 1"
- WCAP-15381, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 2"

C.2 *It appears that discrepancies exist between the end-of-license (EOL) upper-shelf energy (USE) values reported by the licensee and those based on information from reactor vessel integrity database (RVID) for some beltline materials of Byron and Braidwood units. Provide detailed calculations for the EOL (32 effective full power years (EFPY)) USE values for the following material: intermediate to lower shell forging circumferential weld WF336 (heat 442002) for Byron, Unit 1; intermediate to lower shell forging circumferential weld WF447 (heat 442002) and nozzle shell to intermediate shell forging circumferential weld WF562 (heat 442001) for Byron, Unit 2; nozzle shell forging 5P-7016 and intermediate to lower shell forging circumferential weld WF562 (heat 442011) for Braidwood, Unit 1; and nozzle shell forging 5P-7056 and intermediate to lower shell forging circumferential weld WF562 (heat 442011) for Braidwood, Unit 2. The information submitted in the calculations for these materials should include initial USE, chemistry data, 1/4T fluence data, percent decrease in USE, method for calculating the change in USE, and the basis for any discrepancies with the corresponding USE data for these materials in the RVID.*

<u>C.2 Response</u>

This response is provided in sections, one section corresponding to each of the welds listed in the RAI above. Discussions are also supplied relating to potential changes in the RVID.

C.2.a Provide detailed calculations for the EOL (32 EFPY) USE values for the intermediate to lower shell circumferential weld WF336 (Heat # 442002) for Byron Unit 1.

Per WCAP-15391, the EOL (32 EFPY) clad/base metal interface fluence is 1.94×10^{19} n/cm^2 (E > 1.0 MeV).

Per WCAP-15391, the vessel beltline thickness is 8.5 inches.

The ¼T fluence is 1.16×10^{19} n/cm^2 (E > 1.0 MeV)
$= [1.94 \times 10^{19}\ n/cm^2] * e^{(-0.24 * .25 * 8.5)}$).

Per WCAP-15123, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program", the measured % decrease of the surveillance weld (Heat # 442002) is:

Capsule	Fluence	Measured % Decrease
U	0.404×10^{19} n/cm^2 (E > 1.0 MeV)	4%
X	1.57×10^{19} n/cm^2 (E > 1.0 MeV)	12%
W	2.43×10^{19} n/cm^2 (E > 1.0 MeV)	0%

The measured data was plotted on Figure 2 of Regulatory Guide 1.99, Revision 2, and a line parallel to the existing lines was drawn as described in Regulatory Guide 1.99, Revision 2.

The parallel line along with the EOL ¼T fluence was used to determine a predicted % decrease at EOL (32 EFPY).

Surveillance data was used to determine the projected EOL USE. However, for completeness, per WCAP-15391, the best estimate copper and nickel weight percent values for weld heat # 442002 are 0.04 % Copper and 0.63 % nickel.

From Figure 2 of Regulatory Guide 1.99, Revision 2, as modified above, the predicted % decrease is 11%.

Hence, the EOL (32 EFPY) USE is determined as follows:

Initial USE * % decrease + Initial USE = [(77 ft-lb) * (-0.11)] + 77 ft-lb =
68.53 ft-lb ~ 69 ft-lb.

This information will require a revision to RVID. For definition of initial USE, see Section C.2.h.

C.2.b *Provide detailed calculations for the end-of-license (32 EFPY) USE values for the intermediate to lower shell circumferential weld WF447 (Heat # 442002) for Byron Unit 2)*

Per WCAP-15392, the EOL (32 EFPY) clad/base metal interface fluence is 2.03×10^{19} n/cm^2 (E > 1.0 MeV).

Per WCAP-15392, the vessel beltline thickness is 8.5 inches.

The ¼T fluence is 1.22×10^{19} n/cm^2 (E > 1.0 MeV)
$= [2.03 \times 10^{19}\ \text{n/cm}^2] * e^{(-0.24 * .25 * 8.5)}$.

Per WCAP-15176, Revision 0, "Analysis of Capsule X from Commonwealth Edison Company Byron Unit 2 Reactor Vessel Radiation Surveillance Program", the measured % decrease of the surveillance weld (Heat # 442002) is:

Capsule	Fluence	Measured % Decrease
U	0.405×10^{19} n/cm^2 (E > 1.0 MeV)	0%
W	1.27×10^{19} n/cm^2 (E > 1.0 MeV)	0%
X	2.30×10^{19} n/cm^2 (E > 1.0 MeV)	1%

The measured data was plotted on Figure 2 of Regulatory Guide 1.99, Revision 2, and a line parallel to the existing lines was drawn as described in Regulatory Guide 1.99, Revision 2.

The parallel line along with the EOL ¼T fluence was used to determine a predicted % decrease at EOL (32 EFPY).

Surveillance data was used to determine the projected EOL USE. However, for completeness, per WCAP-15392, the best estimate copper and nickel weight percent values for weld heat # 442002 are 0.04 % Copper and 0.63 % nickel.

From Figure 2 of Regulatory Guide 1.99, Revision 2, as modified above, the predicted % decrease is less than 2%.

Hence, the EOL (32 EFPY) USE is determined as follows:

Initial USE * % decrease + Initial USE = [(80 ft-lb) * (-0.02)] + 80 ft-lb =
78.4 ft-lb ~ 78 ft-lb.

This information will require a revision to RVID. For definition of initial USE, see Section C.2.i.

C.2.c Provide detailed calculations for the end-of-license (32 EFPY) USE values for the nozzle to intermediate shell circumferential weld WF562 (Heat # 442011) for Byron Unit 2)

Per WCAP-15392, the EOL (32 EFPY) clad/base metal interface fluence is 0.522×10^{19} n/cm^2 (E>1.0 MeV).

Per WCAP-15392, the vessel beltline thickness is 8.5 inches.

The ¼T fluence is 0.313×10^{19} n/cm^2 (E > 1.0 MeV)
$= [0.522 \times 10^{19} \text{ n/cm}^2] * e^{(-0.24 * .25 * 8.5)}$.

Per WCAP-15389, Revision 0, "Evaluation of Pressurized Thermal Shock for Byron Unit 2", the best estimate weight % of copper (Cu) and nickel (Ni) for circumferential weld WF562 (Heat # 442011) are 0.03 % Cu and 0.67 % Ni.

Utilizing the line for 0.5% Cu and the ¼T fluence the percent decrease was read from Figure 2 of Regulatory Guide 1.99, Revision 2.

From Figure 2 of Regulatory Guide 1.99, Revision 2, the predicted % decrease is 14%.

Hence, the EOL (32 EFPY) USE is determined as follows:

Initial USE * % decrease + Initial USE = [(80 ft-lb) * (-0.14)] + 80 ft-lb =
68.8 ft-lb ~ 69 ft-lb.

This information will require a revision to RVID. For definition of initial USE, see Section C.2.j.

C.2.d *Provide detailed calculations for the end-of-license (32 EFPY) USE values for the nozzle forging 5P-7016 for Braidwood Unit 1*

Per WCAP-15364, the EOL (32 EFPY) clad/base metal interface fluence 6.08×10^{18} n/cm^2 (E > 1.0 MeV).

Per WCAP-15364, the vessel beltline thickness is 8.5 inches.

The ¼T fluence is 3.65×10^{18} n/cm^2 (E > 1.0 MeV)
= [6.08×10^{18} n/cm^2] * $e^{(-0.24 * .25 * 8.5)}$.

Per WCAP-15364, the best estimate weight % of copper (Cu) and Nickel (Ni) for nozzle shell forging 5P-7016 is 0.04 % Cu and 0.73 % Ni.

Utilizing the line for 0.10% Cu and the ¼T fluence, the percent decrease was read from Figure 2 of Regulatory Guide 1.99, Revision 2.

From Figure 2 of Regulatory Guide 1.99, Revision 2, the predicted % decrease is 15%.

Hence, the EOL (32 EFPY) USE is determined as follows:

Initial USE * % decrease + Initial USE = [(155 ft-lb) * (-0.15)] + 155 ft-lb = 131.75 ft-lb ~ 132 ft-lb.

This information will require a revision to RVID. For definition of initial USE, see Section C.2.k.

C.2.e *Provide detailed calculations for the EOL (32 EFPY) USE values for the intermediate to lower shell circumferential weld WF562 (Heat # 442011) for Braidwood Unit 1)*

Per WCAP-15364, the EOL (32 EFPY) clad/base metal interface fluence is 1.99×10^{19} n/cm^2 (E > 1.0 MeV).

Per WCAP-15364, the vessel beltline thickness is 8.5 inches.

The ¼T fluence is 1.19×10^{19} n/cm^2 (E > 1.0 MeV)
= [1.99×10^{19} n/cm^2] * $e^{(-0.24 * .25 * 8.5)}$.

Per WCAP-15316, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 1 Reactor Vessel Radiation Surveillance Program", the measured % decrease of the surveillance weld (Heat # 442011) is:

Capsule	Fluence	Measured % Decrease
U	0.387×10^{19} n/cm^2 (E > 1.0 MeV)	0%
X	1.25×10^{19} n/cm^2 (E > 1.0 MeV)	0%
W	2.09×10^{19} n/cm^2 (E > 1.0 MeV)	7%

The measured data was plotted on Figure 2 of Regulatory Guide 1.99, Revision 2, and a line parallel to the existing lines was drawn as described in Regulatory Guide 1.99, Revision 2.

The parallel line along with the EOL ¼T fluence was used to determine a predicted % decrease at EOL (32 EFPY).

Surveillance data was used to determine the projected EOL USE. However, for completeness, per WCAP-15364, the best estimate copper and nickel weight percent values for weld heat # 442011 are 0.03 % Copper and 0.67 % nickel.

From Figure 2 of Regulatory Guide 1.99, Revision 2, as modified above, the predicted % decrease is 6.3%.

Hence, the EOL (32 EFPY) USE is determined as follows:

Initial USE * % decrease + Initial USE = [(80 ft-lb) * (-0.063)] + 80 ft-lb =
74.96 ft-lb ~ 75 ft-lb.

This information will require a revision to RVID. For definition of initial USE, see Section C.2.I.

C.2.f Provide detailed calculations for the end-of-license (32 EFPY) USE values for the nozzle forging 5P-7056 for Braidwood Unit 2)

Per WCAP-15373, the EOL (32 EFPY) clad/base metal interface fluence is 5.67×10^{18} n/cm^2 (E > 1.0 MeV).

Per WCAP-15373, the vessel beltline thickness is 8.5 inches.

The ¼T fluence is 3.40×10^{18} n/cm^2 (E > 1.0 MeV)
$= [5.67 \times 10^{18}\ \text{n/cm}^2] * e^{(-0.24 * .25 * 8.5)}$.

Per WCAP-15381, Revision 0, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 2", the best estimate weight % of copper (Cu) and nickel (Ni) for nozzle shell forging 5P-7056 is 0.04 % Cu and 0.90 % Ni.

Utilizing the line for 0.10% Cu and the ¼T fluence the percent decrease was read from Figure 2 of Regulatory Guide 1.99, Revision 2.

From Figure 2 of Regulatory Guide 1.99, Revision 2, the predicted % decrease is 15%.

Hence, the EOL (32 EFPY) USE is determined as follows:

Initial USE * % decrease + Initial USE = [(115 ft-lb) * (-0.15)] + 115 ft-lb = 97.75 ft-lb ~ 98 ft-lb.

This information will require a revision to RVID. For definition of initial USE, see Section C.2.m.

C.2.g Provide detailed calculations for the end-of-license (32 EFPY) USE values for the intermediate to lower shell circumferential weld WF562 (Heat # 442011) for Braidwood Unit 2)

Per WCAP-15373, the EOL (32 EFPY) clad/base metal interface fluence is 1.89×10^{19} n/cm^2 (E > 1.0 MeV).

Per WCAP-15373, the vessel beltline thickness is 8.5 inches.

The ¼T fluence is 1.13×10^{19} n/cm^2 (E > 1.0 MeV)
$= [1.89 \times 10^{19}$ n/cm$^2] * e^{(-0.24 * .25 * 8.5)}$.

Per WCAP-15369, Revision 0, "Analysis of Capsule W from Commonwealth Edison Company Braidwood Unit 2 Reactor Vessel Radiation Surveillance Program", the measured % decrease of the surveillance weld (Heat # 442011) is:

Capsule	Fluence	Measured % Decrease
U	0.400×10^{19} n/cm^2 (E > 1.0 MeV)	13%
X	1.23×10^{19} n/cm^2 (E > 1.0 MeV)	1%
W	2.25×10^{19} n/cm^2 (E > 1.0 MeV)	0%

The measured data was plotted on Figure 2 of Regulatory Guide 1.99, Revision 2, and a line parallel to the existing lines was drawn as described in Regulatory Guide 1.99, Revision 2.

The parallel line along with the EOL ¼T fluence was used to determine a predicted % decrease at EOL (32 EFPY).

Surveillance data was used to determine the projected EOL USE. However, for completeness, per WCAP-15373, the best estimate copper and nickel weight percent values for weld heat # 442011 are 0.03 % Copper and 0.67 % nickel.

From Figure 2 of Regulatory Guide 1.99, Revision 2, as modified above, the predicted % decrease is 16%.

Hence, the EOL (32 EFPY) USE is determined as follows:

Initial USE * % decrease + Initial USE = [(80 ft-lb) * (-0.16)] + 80 ft-lb =
67.2 ft-lb ~ 67 ft-lb.

This information will require a revision to RVID. For definition of initial USE, see Section C.2.n.

RVID Changes Due to Evaluation

C.2.h Byron Unit 1: Weld WF-336, Heat # 442002, Flux Type LINDE 80, Flux Lot # 8873

Unirradiated USE:	RVID = 74 ft-lb	Revaluation = 77 ft-lb

The 77 ft-lb value is based on Charpy data given on the "Record of Filler Wire Qualification Test" provided to W by B&W May 6, 1974 (Test Number WF 336). This value is based on a set of three Charpy tests conducted at 200 F and each test resulting in 100% shear. The data used is as follows:

Test Temp. (°F)	Measured Energy (ft-lb)
200	72
200	79
200	81

USE @ EOL (1/4T):	RVID = 64.7 ft-lb	Reevaluation = 69 ft-lb
% Drop @ EOL (1/4T):	RVID = 12.6%	Reevaluation = 11.0%
Fluence (1/4T) @ EOL:	RVID = 1.171	Reevaluation = 1.16

The fluence values used in this evaluation were obtained from WCAP-15391 and include the uprating projections.

C.2.i Byron Unit 2: Weld WF-447, Heat # 442002, Flux Type LINDE 80, Flux Lot # 8064:

Unirradiated USE: RVID = 67 ft-lb Revaluation = 80 ft-lb

The 80 ft-lb value is based on Charpy data given on the "Record of Filler Wire Qualification Test" provided to W by B&W August 8, 1974 (Test Number WF 447). This value is based on a set of three Charpy tests conducted at 150 F and each test resulting in 100% shear. The data used is as follows:

Test Temp. (°F)	Measured Energy (ft-lb)
150	81
150	83
150	77

USE @ EOL (1/4T):	RVID = 58.5 ft-lb	Reevaluation = 78 ft-lb
% Drop @ EOL (1/4T):	RVID = 12.6%	Reevaluation = 2.0%
Fluence (1/4T) @ EOL:	RVID = 1.177	Reevaluation = 1.22

The fluence values used in this evaluation were obtained from WCAP-15392 and include the uprating projections.

C.2.j Byron Unit 2: Weld WF-562, Heat # 442011, Flux Type LINDE 80, Flux Lot # 8064:

Unirradiated USE: RVID = 70 ft-lb Revaluation = 80 ft-lb

The 80 ft-lb value is based on Charpy data given on the "Record of Filler Wire Qualification Test" provided to W by B&W August 11, 1975 (Test Number WF 562). This value is based on a set of three Charpy tests conducted at 250 F and each test resulting in 100% shear. The data used is as follows:

Test Temp. (°F)	Measured Energy (ft-lb)
250	82
250	81
250	76

USE @ EOL (1/4T):	RVID = 61.7 ft-lb	Reevaluation = 69 ft-lb
% Drop @ EOL (1/4T):	RVID = 11.9%	Reevaluation = 14.0%
Fluence (1/4T) @ EOL:	RVID = 0.303	Reevaluation = 0.313

The fluence values used in this evaluation were obtained from WCAP-15392 and include the uprating projections.

ATTACHMENT 1
(continued)

C.2.k Braidwood Unit 1: Nozzle Shell Forging 5P-7016

Unirradiated USE: RVID = 162 ft-lb Revaluation = 155 ft-lb

The 155 ft-lb value is based on Charpy data given on the "Material Analysis Report" provided to W by B&W April 7, 1975 (Part # 157982C-1). This value is based on a set of three axial orientated Charpy tests conducted at 150°F. The data used is as follows:

Test Temp. (°F)	Measured Energy (ft-lb)
150	155.5
150	141
150	170

USE @ EOL (1/4T):	RVID = 136.9 ft-lb	Reevaluation = 132 ft-lb
% Drop @ EOL (1/4T):	RVID = 14.3%	Reevaluation = 15.0%
Fluence (1/4T) @ EOL:	RVID = 0.299	Reevaluation = 0.365

The fluence values used in this evaluation were obtained from WCAP-15364 and include the uprating projections.

Limiting Material: RVID = YES Reevaluation = NO

Per WCAP-15365 intermediate to lower shell weld WF 562 is limiting.

C.2.I Braidwood Unit 1: Weld WF-562, Heat # 442011, Flux Type LINDE 80, Flux Lot # 8061:

Unirradiated USE: RVID = 70 ft-lb Revaluation = 80 ft-lb

The 80 ft-lb value is based on Charpy data given on the "Record of Filler Wire Qualification Test" provided to W by B&W August 11, 1975 (Test Number WF 562). This value is based on a set of three Charpy tests conducted at 250 F and each test resulting in 100% shear. The data used is as follows:

Test Temp. (°F)	Measured Energy (ft-lb)
250	82
250	81
250	76

USE @ EOL (1/4T):	RVID = 58.2 ft-lb	Reevaluation = 75 ft-lb
% Drop @ EOL (1/4T):	RVID = 16.9%	Reevaluation = 6.3%
Fluence (1/4T) @ EOL:	RVID = 1.345	Reevaluation = 1.19

The fluence values used in this evaluation were obtained from WCAP-15364 and include the uprating projections.

C.2.m Braidwood Unit 2: Nozzle Shell Forging 5P-7056

Unirradiated USE: RVID = 128 ft-lb Revaluation = 115 ft-lb

The 115 ft-lb value is based on Charpy data given on the "Material Analysis Report" provided to W by B&W May 22, 1975 (Contract # 640-0016—51-24). This value is based on a set of three axial orientated Charpy tests conducted at 150 F. The data used is as follows:

Test Temp. (°F)	Measured Energy (ft-lb)
150	128
150	103
150	115

USE @ EOL (1/4T):	RVID = 110.3 ft-lb	Reevaluation = 98 ft-lb
% Drop @ EOL (1/4T):	RVID = 13.8%	Reevaluation = 15.0%

Fluence (1/4T) @ EOL: RVID = 0.261 Reevaluation = 0.340

The fluence values used in this evaluation were obtained from WCAP-15373 and include the uprating projections.

Limiting Material: RVID = YES Reevaluation = NO

Per WCAP-15381 intermediate to lower shell weld WF 562 is limiting.

C.2.n Braidwood Unit 2: Weld WF-562, Heat # 442011, Flux Type LINDE 80, Flux Lot # 8061:

Unirradiated USE: RVID = 70 ft-lb Revaluation = 80 ft-lb

The 80 ft-lb value is based on Charpy data given on the "Record of Filler Wire Qualification Test" provided to W by B&W July 11, 1977 (Test Number WF 562). This value is based on a set of three Charpy tests conducted at 250 F and each test resulting in 100% shear. The data used is as follows:

Test Temp. (°F)	Measured Energy (ft-lb)
250	82
250	81
250	76

USE @ EOL (1/4T):	RVID = 58.2 ft-lb	Reevaluation = 67 ft-lb
% Drop @ EOL (1/4T):	RVID = 16.8%	Reevaluation = 16.0
Fluence (1/4T) @ EOL:	RVID = 1.320	Reevaluation = 1.13

The fluence values used in this evaluation were obtained from WCAP-15373 and include the uprating projections.

C.3 *In Section 5.1 of the Power Uprate Safety Evaluation, the licensee indicated that the methodology used to generate P-T limit curves for the Byron and Braidwood units is the methodology documented in WCAP-14040-NP-A, as modified by five alternatives (exceptions) to the methodology. Acceptance of the Pressure-Temperature Limits Report (PTLR) was predicated on acceptance of the methodology of WCAP-14040-NP-A, as modified by the methods of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Case N-514 for establishing low temperature overpressure (LTOP) system setpoints. The safety evaluation for acceptance of the license amendment for the PTLR never included or accepted the five alternatives as the basis for generating the P-T limit curves for the PTLR. With respect to these five alternatives, the following actions are necessary:*

a. *Since the alternatives to the methodology will change the previous methodology initially approved for the Byron/Braidwood PTLR, pursuant to the staff's position stated in Generic Letter (GL) 96-03 for accepting PTLR requests, the licensee will need to submit a license amendment request to change the appropriate administrative control section for the PTLR in the technical specifications to incorporate any proposed alternatives to the previously approved methodology, and submit these five alternative methods for review and approval. In this case, it should be noted that the staff, at this time, is not approving any proposal to eliminate the flange temperatures requirements, because the staff's review of WCAP-15315 is still pending.*

C.3.a Response

As noted above, the new heatup and cooldown curves were generated using the NRC approved methodology documented in WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," with the following exceptions:

1. The fluence values used in this report are calculated fluence values, not the best estimate fluence values.
2. The K_{Ic} critical stress intensities are used in place of K_{Ia} critical stress intensities. This methodology is taken from approved ASME Code Case N-640.
3. The reactor vessel flange temperature requirement has been eliminated. Justification has been provided in WCAP-15315.
4. The 1996 Version of Appendix G to Section XI was used rather than the 1989 version.
5. The methodology from approved ASME Code Case N-588 was used to consider circumferentially oriented flaws in development of P-T Curves.

Exceptions 1, 2, 3, and 5 will be addressed in a near term license amendment request which will be transmitted to the NRC independent of this response. Specifically, Exceptions 2, 3, and 5 will be addressed as noted under C.3.b below. Exception 4 has already

been approved by the NRC as documented in a Safety Evaluation transmitted by letter from R. A. Capra (USNRC) to O. D. Kingsley (Commonwealth Edison Company), "Byron Station Units 1 and 2, and Braidwood Station, Units 1 and 2, Acceptance for Referencing of Pressure Temperature Limits Report," dated January 21, 1998.

b. *Any proposals to modify the existing PTLR methodology by the methods of Code Case N-588 or N-640 will require, pursuant to 10 CFR 50.60(b), exemption requests to deviate from the requirements of Section IV.A.2 of Appendix G to 10 CFR Part 50. The staff will evaluate such exemptions on a case-by-case basis pursuant to the exemption approval criteria stated in 10 CFR 50.12. In this case, any proposal to use Code Case N-640 must be accompanied with a statement that the provisions in Code Case N-514 (or in Paragraph G-2215 of the 1996 Edition of Appendix G to the Code) for establishing the LTOP pressure setpoint at 110 percent of the allowable pressure provided by the P-T limit curves for normal operation will not be used, and that instead, the LTOP setpoint will be established at 100 percent of the allowable pressure established by the P-T limit curves for normal operation.*

C.3.b Response

A request for an exemption to 10 CFR 50.60, "Acceptance criteria for fracture prevention measures for lightwater nuclear power reactors for normal operation," will be submitted in the near term to address exceptions 2, 3, and 5.

NRC Question Set D

D.1 *Summarize the contents and results of the assessment that was used to confirm that the current steam generator (SG) plugging criteria in the Technical Specifications for each unit will remain adequate for the power uprated conditions (Section 5.7).*

D.1 Response

An evaluation was performed to assess the impact of the uprated parameters on the tube repair criteria. Regulatory Guide 1.121 provides guidance for the determination of a repair limit for SG tubes undergoing localized tube wall thinning. Based on the conservative assumption of uniform wall thinning over an unlimited length, the resulting structural limit $[(t_{nom} - t_{min})/t_{nom}]$, for the free-span region of the tube, for the Byron Unit 1 and Braidwood Unit 1 steam generators is 60.0%; and for the Byron Unit 2 and Braidwood Unit 2 steam generators is 58.1%. These limiting values are based on low T_{avg} transient conditions and the American Society of Mechanical Engineers (ASME) Code minimum properties. As recommended in paragraph C.2.b of the Regulatory Guide, an additional thickness degradation allowance must be added to the structural limit to establish the tube repair limit. Paragraph C.3.f of the Regulatory Guide specifies that the

the operational degradation allowance include continuing degradation and consideration of eddy current measurement errors.

Power uprate will result in no changes to the eddy current measurement errors and changes to continuing degradation growth rates have been identified for evaluation of the Technical Specification Repair Limit. Sufficient repair limit allowances exist for eddy current measurement uncertainty and continuing degradation growth at the power uprate conditions to ensure that the tubing structural limit is not exceeded.

For steam generator tubes repaired using sleeve technology, a change to the Technical Specification repair limit would be required to maintain the existing 20% allowance for the operational degradation allowance. There are presently no sleeves installed in Byron and Braidwood Unit 1 and 2 and therefore no operational experience with flaw growth rates.

D.2 *For the Byron, Unit 2, and Braidwood, Unit 2, SGs, please provide a summary of the operational assessment for antivibration bar (AVB) wear due to the power uprate to determine the allowable operating interval between inspections (Section 5.7.2.2.5).*

<u>D.2 Response</u>

Operational Assessments have been performed for AVB wear using the guidelines of EPRI Steam Generator Integrity Assessment Guidelines Revision 1. The Operational Assessments consider the largest flaw left in service based upon a 40% TW repair limit, NDE uncertainties associated with the EPRI NDE technique qualification and analyst capabilities and unit specific growth rates adjusted by a conservative uprate wear factor. The limiting end of cycle flaw condition is assessed against the structural limit calculated consistent with the guidelines of Regulatory Guide 1.121 updated for power uprate conditions. The structural limit is calculated using ASME Code minimum material properties. The probability of detection for AVB wear at Byron and Braidwood Unit 2 is good as documented in the EPRI technique performance and site performance demonstrations.

The effect of the Uprate on AVB wear on tubes with reported wear, based on changes to the thermal-hydraulic conditions, has been evaluated to conservatively determine the increase in wear at Byron Unit 2 and Braidwood Unit 2. The projected wear rate increase is consistent with what has been seen for other plants with similar uprates.

Active tubes for both Units, currently with no reported wear indications, have been in service for more than 10 years and either have no AVB wear as a result of thermal-hydraulic induced forces, or exhibit such small wear rates, that the effect of the Uprate is not significant. The incidence of new wear sites as a result of the Uprate is expected to be small.

In summary Operational Assessments have been performed that demonstrate performance criteria are satisfied for the inspection interval considering uprate

conditions. The Operational Assessments will be updated to reflect any planned inspections performed prior to implementing the power uprate.

D.3 *Page 5-104 states that, "The window for operating conditions at 5 percent power uprate (3600.6MWt) is the same as, or bounded by the window for, the operating conditions (3425 MWt) previously evaluated. For example, the maximum primary side T_{hot} (most important parameter with respect to tube degradation) is 618.4°F for both 3425 MWt and 3600.6 MWt." Page 5-124 states that, "To minimize the potential impact of the power uprate on SG tube degradation, selection of the optimum point (best estimate steam pressure) at which to design the high pressure turbine modifications includes maintaining the post-uprate T_{hot} at the same value as the current T_{hot}...." Because this information is inconsistent, please clarify if T_{hot} indeed stays the same after the uprate. If T_{hot} is increased after the power uprate, what effect will this have on the discussion in Section 5.7.2.4 on tube degradation.*

D.3 Response

Section 5.7.1.4 on page 5-104 introduces the discussion of Tube Degradation for the Unit 1 Babcock and Wilcox Replacement Steam Generators. The purpose of this paragraph is to establish that the maximum design limit for T_{hot} pre-uprate was 618.4°F and it remains the maximum design limit post-uprate. The unit 1 RSGs will be operated post-uprate at a T_{hot} of approximately 617°F.

Section 5.7.2.4 on page 5-124 addressed tube degradation for the Unit 2 Westinghouse Steam Generators. It states that the post-uprate T_{hot} will be maintained at the same values as the current T_{hot}. The T_{hot} limit of 611°F, established as a result of the T_{hot} reduction program will indeed be maintained post-uprate. However, the analysis performed to support power uprate is bounding up to a T_{hot} of 618.4°F and ComEd may in the future choose to revise the T_{hot} limit. In that case the impact of a revised T_{hot} on tube degradation would be evaluated.

NRC Question Set E

E.1 *Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will new operator actions be required?*

The licensee stated in its letter dated July 5, 2000 (page A-21) that, "The power uprate has the potential to affect plant procedures used to operate and maintain the facility in accordance with design basis and licensing requirements.... Procedures that are identified as being affected by the power uprate will be revised prior to the uprate implementation." In Attachment E (page 12-1), the licensee stated that, "A physical review of each procedure identified [during the screening] will be conducted to determine the need for revision. Those procedures will be revised to incorporate changes. For example, changes due to modifications, operator response times, setpoint changes will result in revisions

to existing procedures... and all required training will be conducted prior to the implementation of the power uprate."

In addition to the information provided by the licensee in its July 5, 2000, submittal, what procedures will be changed, what changes will be made and, will new operator actions be required?

E.1 Response

The major procedure impact due to the power uprate is on the Appendix J surveillance procedures. These procedures will be revised to reflect the post-accident peak containment pressure (P_a) and allowable leakage (L_a) identified in the proposed Technical Specification change in support of power uprate. The power uprate did not "require" changes to the scope or nature of operator actions; however, one response time was changed as noted in the response to E.2 below. It did, however, cause range changes to process parameters (e.g., pressures, temperatures, and flows), that will result in several procedures being changed to reflect the uprated parameters. The process parameter changes will be reflected in changes to normal operating, engineering, surveillance, and emergency operating procedures.

E.2 *Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will effect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. If there are any reduced operator response times, discuss why they are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to complete the required manual actions in the times allowed. Discuss results of simulator exercises conducted to assure that operator response times for operator actions that are potentially sensitive to power uprate, can be successfully achieved within allowable time limits.*

E.2 Response

Results of the accident analyses showed that required operator response times are not affected by power uprate; however, as part of the activities performed in support of power uprate, the assumed operator response time to isolate auxiliary feedwater to the faulted steam generator during a feedwater line break event was administratively reduced from 30 to 20 minutes. This change was made to establish consistency with the assumed operator response time for the steamline break outside containment. Both of these events direct the operators to the same emergency response procedures; therefore, making the response time consistent is beneficial from a human factors standpoint.

E.3 *Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g., normal range, marginal range, and out-or-tolerance range)? If changes will occur, discuss how they will be addressed.*

In Attachment C (page C-24) of the licensee's July 5, 2000, submittal, the licensee indicates that, "The basic design of all systems remains unchanged and no new equipment or systems have been installed which could potentially introduce new failure modes or accident sequences. No changes have been made to any reactor trip or ESF actuation setpoints." However, the licensee also stated that, "Minor modifications, to support implementation of uprated power conditions, will be made as required to existing systems and components." In addition to addressing the specific questions in question E.3, what are the "minor modifications" referred to by the licensee and what effect will they have on operator performance?

E.3 Response

Station instrumentation setpoint/scaling adjustments are being performed within the 7300 control system at both Byron and Braidwood Stations. Adjustments for the new operating Tavg and ΔT are being incorporated. Adjustments to the turbine driven feedwater pumps (TDFWP) speed control systems will also be performed. Any annunciator alarm points or computer points identified during the development of the setpoint/scaling changes are being reviewed for impact of the adjustment. Alarm/computer adjustments will then be performed during the setpoint/scaling change. Any computer points not covered by the setpoint/scaling changes (i.e. thermal power) will be revised during the uprate implementation. These will be identified via an engineering review of all computer points. Identified computer points will also be assessed for impact on station displays.

All meter scale adjustments, not covered under an already identified instrument setpoint/scaling adjustment, will be performed under the stations Design Change Process (DCP). The DCP process requires review of the change by the impacted departments. The station Operating and Training Departments will assess the impact on the operating staff. Revisions to the operator-training program are then incorporated and training disseminated.

All zone-banding changes will be documented and submitted for Plant Review. Station Engineering Request and Maintenance Work Requests will be generated for procurement of the scales and installation tracking. All banding changes will be in accordance with the ComEd Human Factors Manual and appropriate station procedures.

The minor changes being performed at Byron and Braidwood Stations include the following Instrumentation and Control setpoint/scaling adjustments: LTOP, ΔT, $f_1(\Delta I)$, 1st stage impulse pressure, load rejection controller. Refer to the response for question B.2 for further description of the changes. I&C component changes involve replacing the Unit 1 first stage pressure transmitters with an instrument that can span the increased steam pressure range. No other I&C modifications will be performed for the uprate. Mechanical component changes involve valve replacements and valve trim changes to increase the valve flow coefficient (i.e., C_v) for several heater drain valves and the addition of gland

steam piping to increase the high pressure turbine gland leak-off removal capacity.

E.4 *Discuss all changes the power uprate will have on the Safety Parameter Display System (SPDS) and how they will be addressed.*

Response

No changes will be made to the process parameters that input to SPDS and no physical changes will be made to the SPDS display from a human factors perspective.

The process computer points are being reviewed and updated as required to support implementation of the power uprate. The field data computer points associated with SPDS will be rescaled as necessary and appropriate revisions to the software made prior to implementation of power uprate. The review and identification of required changes will be completed in accordance with the power uprate implementation schedule. All meter scale adjustments, not covered under an already identified instrument setpoint/scaling adjustment, will be performed under the stations DCP. The DCP process requires review of the change by the impacted departments. The station Operating and Training Departments will assess the impact on the operating staff. Revisions to the operator-training program are then incorporated and training disseminated.

E.5 *Describe all changes the power uprate will have on the operator training program and the plant simulator.*

Response

The appropriate lesson plans will be revised to reflect the new operating conditions due to the power uprate effort. Station operators will be trained on the expected system parameter changes. Operators will also be trained on the impacts on BOP and NSSS system margins. The operators will also be trained on changes to abnormal, emergency and normal operating procedures. Just prior to the actual uprates, operators will receive Heightened Level of Awareness (HLA) training on the uprate program.

Plant modifications resulting from design changes and setpoint/scaling changes, are being incorporated through the Design Change Process (DCP). Changes to the simulator software, to reflect uprated power operations and support operator training, will be implemented prior to the power uprate in accordance with the power uprate implementation schedule. The initial list of simulator changes that have been identified are listed below.

- Revised decay heat load curve
- Revised auxiliary feedwater flow as a function of time
- Revised containment spray system response times
- Containment temperature/pressure post accident conditions
- Containment heat removal performance

- Revised hydrogen generation rates
- Revised safety injection response times
- Revised cooldown curve for normal/post trip shutdown
- Revised RHR heat exchanger loads
- Revised Tavg, Thot, Tcold values
- Revised reactor coolant system scaling
- Revised radiation source terms
- Revised component cooling water system heat loads
- Revised ultimate heat sink loads as a function of time
- Revised spent fuel pool heat loads and pool temperatures
- Revised flow, pressure and temperature data for the following systems:
  - Main Steam
  - Heater Drain
  - Condensate
  - Condensate Booster
  - Main Feedwater
  - Extraction Steam
  - Circulating Water
- Revised turbine impulse pressure and scaling information
- Revised HD control valve Cv curves
- Revised FW Pump speed/dP program scaling
- Revised FW transient response
- Revised WS temperature control valve positions

<u>NRC Question Set F</u>

With regard to radiological analyses, key elements of the staff review are comparisons of radiological consequence analyses discussed in the most recent evaluations of record (i.e., past analyses, which in most cases are expected to be those summarized in the Byron and Braidwood Updated Final Safety Analysis Report (UFSAR)) with the corresponding consequence analyses discussed in the proposed license amendment (i.e., current analyses, which are intended to support the proposed power up-rate). Specifically, for each postulated event or condition, the staff is interested in comparing and understanding the differences between past and current estimates of: (a) the concentrations of iodine and noble gas fission products available for release, (b) the concentrations of iodine and noble gases released to the environment as a result of the event or condition, as well as (c) the associated doses at the nearest exclusion area boundary (EAB), the low population zone (LPZ) outer boundary, and the control room. Therefore, please provide answers to the following:

F.1 *What dose models were used for each of the accidents reanalyzed to support power uprate? Have the computer codes and methods of analysis used been approved by the NRC? If they have not been approved, please provide the basis for their use.*

F.1 Response

As noted in the Power Uprate Licensing Report, Section 6.7.1.3, "Computer Code," on page 6-631, the TITAN5 computer code was used for all Chapter 6 radiological dose calculations, except for the Steam Generator Tube Rupture (SGTR) analysis as noted below. TITAN5 is a long-standing Westinghouse Electric Company code which has been previously used for Byron and Braidwood Stations licensing activities. Although not formally approved by the NRC, the NRC has performed confirmatory analysis on numerous applications and has obtained results consistent with the TITAN5 results. The RETRAN computer code was used to perform thermal hydraulic and activity release calculations for the SGTR analysis, consistent with the previously NRC approved SGTR analysis for Byron and Braidwood Stations as documented in the following letters: 1) letter from G. F. Dick (NRC) to O. D. Kingsley (ComEd), "Revised Steam Generator Tube Rupture Analysis – Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated January 28, 1998; 2) letter from G. F. Dick (NRC) to O. D. Kingsley (ComEd), "Revised Steam Generator Tube Rupture Analysis – Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2," dated March 11, 1998; and 3) letter from S. N. Bailey (NRC) to O. D. Kingsley (ComEd), "Revised Steam Generator Tube Rupture Analysis – Byron Unit 2, and Braidwood Unit 2," dated May 25, 1999

F.2 *What models, codes, and methods of analysis were used previously, i.e., in the initial licensing of Byron and Braidwood or the analysis of record (UFSAR)?*

F.2 Response

For the initial licensing of Byron and Braidwood Stations, Westinghouse used TITAN5 to generate the radiological release data. Sargent & Lundy then took this data and developed the dose numbers using hand calculations, as described in UFSAR Attachment 15A, "Dose Models Unsed to Evaluate the Environmental Consequesces of Accidents," with the exception of the Control Room dose calculation. The Control Room dose was calculated using the Sargent and Lundy computer code "POST-DBA."

F.3 *All reanalyses for radiological consequences for the design basis accidents (discussed in Chapter 6) include the 2 percent instrument error margin; please provide the basis for not using the instrument error margin in the reanalysis of normal operation dose rates, shielding, and annual radwaste effluent release (discussed in Chapter 9). It should be noted that Regulatory Guide 1.49, which the licensee quotes in Chapter 9, specifically states that, "analyses and evaluation in support of the application should be made at an assumed core power level equal to 1.02 times the proposed licensed power level... for (a) normal operating conditions... (c) accident conditions...."*

F.3 Response

As noted in Appendix A of the Byron Station and Braidwood Station UFSAR, the extent of commitment to the recommendations of Regulatory Guide (RG) 1.49, Revision 1 is limited for Byron and Braidwood Stations for operation at a power

level less than 3800 MWt, and utilization analyses and evaluations based on core power levels less than the power levels of the guide.

The Byron Station and Braidwood Station power uprate assessment elected to specifically address the RG 1.49 recommended 2% margin for post accident offsite radiological consequences, since it is clear from the reading of the Introduction Section of RG 1.49, that other than fixing a maximum allowable power level for all nuclear plants, the focus of this guide is to establish a margin requirement on power level for evaluations of "major structures, systems, and components of the facility which bear significantly on the acceptability of the site evaluation factors identified in 10CFR Part 100".

In addition, as part of the power uprate assessment, this margin was addressed, not only for site boundary dose analyses, but for all post accident radiological evaluations including the post-accident component of the equipment qualification dose, and post-accident shielding/vital access. However, for the reasons discussed below, Byron Station and Braidwood Station did not include this 2% instrument error margin for normal operation radiological evaluations.

Our understanding of the three Regulatory Positions noted in RG 1.49 as follows.

- Position C.1 requires that the licensed power level of all nuclear power plants should be limited to a core power level of 3800MWt.

- Position C.2 requires that the "thermo-hydraulic" analyses done for a) normal operations; b) transient conditions such as load changes, control rod malfunctions, improper operations, loss of coolant flow, loss of load or turbine trip, etc.; and c) accident conditions necessary to evaluate the adequacy of structures, systems and components provided for prevention of accidents and mitigation of the consequences of accidents, must be based on 1.02 times the proposed power level.

- Position C.3 requires that analyses, of offsite "radiological" consequences of DBAs to demonstrate compliance with 10CFR Part 100, be based on 1.02 times the core power level.

  Position C.2 is applicable only to "thermo-hydraulic" analyses and not "radiological" analyses. If C.2 was applicable to radiological evaluations, Position C.3, which specifically addresses radiological consequences, would be unnecessary, as it would be covered by Item C.2.c. This position is further strengthened by the fact that neither 10 CFR 20 nor 10 CFR 50 Appendix I (i.e., federal regulations applicable to normal operation radiological issues such as exposure to normal operation dose rates and radwaste effluents) are addressed in the RG 1.49.

The above notwithstanding, a discussion regarding the need to address a 2% margin in analyses supporting normal operation dose rates, shielding, radwaste effluents is given below.

The source terms used in determining normal operation dose rates and shielding are already highly conservative as they reflect design basis values of 1% failed fuel and 100% capacity factors. This conservatism, in conjunction with the conservative analytical techniques used, provide sufficient margin to encompass the intent of RG 1.49 relative to margin.

Effluent releases are in general calculated based on average operation/release data from operating plants which can vary considerably between plants. For example, according to NUREG 0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," Revision 1, the nominal reactor coolant /secondary coolant data used to develop effluent releases is applicable over a core power range from 3000 MWt to 3800 MWt which reflects a ± 12% uncertainty in power level. A 2% margin due to instrument error is considered insignificant compared to this wide range of uncertainty in the utilized data.

The following requests for comparisons and discussions are expected to focus on those changes of input data, assumptions, models, and methodology that make a significant contribution to the differences, as well as the rational for the changes. Receipt of this information should minimize staff interactions with the licensee concerning the details of the various specific calculations. To facilitate the comparison we request the following information for each event or condition analyzed:

F.4 *Present and discuss the differences between the equilibrium fission product concentrations used in the analyses of record, i.e., past analyses, and those used in analyses for the proposed up-rated thermal power, i.e., the current analyses.*

F.5 *Present and discuss the differences between the concentrations of iodine and noble gases released to the environment used in past analyses and those used in the current analyses (include discussion of processes and systems which reduce the concentrations, e.g., plate-out, filtration, adsorption).*

F.6 *Present and discuss the differences between the doses calculated in the past analyses with those calculated in the current analyses.*

F.4, F.5, and F.6 Response

Tables F-1 to F-4 shows comparisons of offsite dose results prior to and after uprate for Byron at the Exclusion Area Boundary (EAB) and Low Population Zone (LPZ) and Braidwood at the EAB and LPZ, respectively. Comparison of control room dose prior to and after uprate are provided in Tables F-5 and F-6 for Byron and Braidwood, respectively. Only LOCA control room dose results are presented in Tables F-5 and F-6, consistent with the current licensing basis. Control room dose results for other accidents are discussed in response to question F.7.

The input assumptions for each event are provided in Section 6 of the Power Uprate Licensing Report. Table F-7, which also references data from Tables F-8 through F-15, is a summary of the changes of input data, assumptions, models,

and methodology that make a significant contribution to the differences in radiological consequence prior to and after uprate. In general, the most significant contributors to the differences in radiological consequence at EAB and LPZ prior to and after uprate are the changes in core inventory, Reactor Coolant System (RCS) fission product concentration, thyroid dose conversion factors, and whole body dose conversion factors. Additional changes for each event are discussed below.

For the steamline break event, the current analysis already uses the International Commission on Radiological Protection (ICRP) Publication 30 thyroid dose conversion factors. The increase in faulted SG mass, primary to secondary leakage, spike factor, and spike duration assumptions, combined with the general significant changes discussed above resulted in an increase in consequences for all cases. The increase in LPZ consequence is higher than the increase in EAB consequence primarily due to the change in iodine spike duration.

For the locked rotor event, the increase in primary to secondary leakage assumption, combined with the general significant changes discussed above resulted in an increase in consequences for all cases.

The locked rotor with stuck open PORV event is bounded by the steamline break event prior to uprate. For power uprate, the thermal hydraulic analysis for the locked rotor event resulted in failed fuel while no failed fuel was predicted for the steamline break event. Therefore, the locked rotor with stuck open PORV is no longer bounded by steamline break and an analysis was performed to support power uprate. A bounding failed fuel assumption of 2% is used. This failed fuel assumption resulted in higher consequences than the steamline break event.

For the rod ejection event, the increase in steam releases and fraction of fuel melting assumptions, combined with the general significant changes discussed above resulted in small differences for the thyroid dose results and an increase in whole body dose results after uprate.

For the small line break outside containment event, the increase in iodine partitioning factor, combined with the general significant changes discussed above resulted in small differences in consequences.

For the steam generator tube rupture event pre-accident spike case, the combined effect of the general significant changes discussed above resulted in a decrease in thyroid dose results and small differences in whole body dose results after uprate.

For the steam generator tube rupture event concurrent spike case, the pre-uprate analysis used more conservative values than those provided in Table F-9 (for example, a value of 2.5 μCi/gm was used for I – 131 instead of 0.66). Therefore, the decrease in thyroid dose results is larger than the pre-accident spike case. The differences between the pre-uprate and post-uprate whole body dose results are small.

For the large break LOCA offsite dose cases, the change in spray modeling assumptions as noted in Licensing Report Section 6.7.8, "Large-Break Loss of Coolant Accident," combined with the general significant changes discussed above resulted in a decrease in consequences for the containment leakage cases. For the ECCS recirculation leakage case, the decrease in sump water volume assumption contributed to the increase in consequences.

For the gas decay tank failure event, the decrease in gas decay tank inventory assumption, combined with the change in whole body dose conversion factors resulted in a decrease in consequences.

For the recycle holdup tank failure event, the increase in recycle holdup tank inventory assumption, combined with the change in thyroid and whole body dose conversion factors resulted in an increase in consequences.

For the spent resin tank failure event, the increase in spent resin tank inventory assumption, combined with the change in thyroid and whole body dose conversion factors, resulted in an increase in consequences.

For the fuel handling accident event, the decrease in the "Incore Decay Time" after shutdown assumption from 100 hours to 48 hours, combined with the general significant changes discussed above, resulted in an increase in consequences.

For LOCA control room calculations, the changes in unfiltered inleakage, control room volume, control room air flow rate, and atmospheric dispersion factors assumptions, combined with the changes in core inventory and thyroid dose conversion factors, resulted in the changes in results shown in Tables F-5 and F-6. The trend for Byron and Braidwood are different because the changes in atmospheric dispersion factors assumptions are different for the two stations.

ATTACHMENT 1
(continued)

Table F-1
Comparison of Byron EAB results

	Pre-Uprate		Post-Uprate	
	Thyroid rem	Whole Body rem	Thyroid rem	Whole Body rem
SLB – Preaccident Spike	2.77*	0.00113*	4.8	0.02
SLB – Concurrent Spike	2.22*	0.00113*	6.3	0.05
Locked Rotor	2.36	0.28	4.1	0.4
Locked Rotor-PORV	n/a	n/a	14.3	0.3
Rod Eject	44.3	0.419	42.2	1.0
Small Line Break – Preaccident Spike	n/a	n/a	1.0	0.03
Small Line Break – Concurrent Spike	0.308	0.038	0.3	0.03
*SGTR – Preaccident Spike	19.72	0.21	16.2	0.21
*SGTR – Concurrent Spike	18.46	0.21	10.0	0.26
LB LOCA – Containment Leakage	114	5.26	64	2.6
LB LOCA – ECCS Recirculation	0.927	0.00241	1.6	0.0075
Gas Decay Tank	n/a	1.02	n/a	0.54
Recycle Holdup Tank	0.0099	0.25	0.85	0.44
Spent Resin Tank	0.283	0.000083	0.45	0.00016
FHA	50.3	0.673	55	1.4

* Results from the limiting unit

ATTACHMENT 1
(continued)

Table F-2
Comparison of Byron LPZ results

	Pre-Uprate		Post-Uprate	
	Thyroid rem	Whole Body rem	Thyroid rem	Whole Body rem
SLB – Preaccident Spike	0.295*	0.000131*	0.5	0.002
SLB – Concurrent Spike	0.309*	0.000131*	2.7	0.02
Locked Rotor	0.93	0.018	1.4	0.03
Locked Rotor-PORV	n/a	n/a	0.9	0.02
Rod Eject	3.9	0.0166	4.4	0.05
Small Line Break – Preaccident Spike	n/a	n/a	0.03	0.0008
Small Line Break – Concurrent Spike	n/a	n/a	0.007	0.0007
*SGTR – Preaccident Spike	0.59	0.0063	0.5	0.007
*SGTR – Concurrent Spike	0.55	0.0062	0.36	0.01
LB LOCA – Containment Leakage	15.8	0.342	6.6	0.2
LB LOCA – ECCS Recirculation	0.194	0.000223	0.31	0.00059
Gas Decay Tank	n/a	0.0305	n/a	0.02
Recycle Holdup Tank	0.0003	0.0076	0.026	0.014
Spent Resin Tank	0.0084	0.0000025	0.014	0.0000047
FHA	1.5	0.0201	1.7	0.03

* Results from the limiting unit

ATTACHMENT 1
(continued)

Table F-3
Comparison of Braidwood EAB Results

	Pre-Uprate		Post-Uprate	
	Thyroid rem	Whole Body rem	Thyroid rem	Whole Body rem
SLB – Preaccident Spike	3.84*	0.00153*	6.5	0.03
SLB – Concurrent Spike	3.1*	0.00153*	8.5	0.06
Locked Rotor	3.2	0.38	5.5	0.5
Locked Rotor-PORV	n/a	n/a	19.2	0.3
Rod Eject	59.9	0.566	57.0	1.4
Small Line Break – Preaccident Spike	n/a	n/a	1.4	0.04
Small Line Break – Concurrent Spike	0.417	0.051	0.3	0.04
*SGTR – Preaccident Spike	26.6	0.29	21.9	0.29
*SGTR – Concurrent Spike	24.9	0.29	13.6	0.35
LB LOCA – Containment Leakage	154	7.1	86	3.4
LB LOCA – ECCS Recirculation	1.25	0.00325	2.1	0.011
Gas Decay Tank	n/a	1.38	n/a	0.73
Recycle Holdup Tank	0.00013	0.34	1.2	0.6
Spent Resin Tank	0.38	0.00014	0.61	0.00021
FHA	67.9	0.909	74	1.8

* Results from the limiting unit

Table F-4
Comparison of Braidwood LPZ results

	Pre-Uprate		Post-Uprate	
	Thyroid rem	Whole Body rem	Thyroid rem	Whole Body rem
SLB – Preaccident Spike	1.32*	0.000592*	2.2	0.006
SLB – Concurrent Spike	1.39*	0.000592*	12.2	0.05
Locked Rotor	4.41	0.075	6.7	0.1
Locked Rotor-PORV	n/a	n/a	4.0	0.06
Rod Eject	17.9	0.0707	19.8	0.2
Small Line Break – Preaccident Spike	n/a	n/a	0.2	0.004
Small Line Break – Concurrent Spike	n/a	n/a	0.03	0.003
*SGTR – Preaccident Spike	2.46	0.026	2.09	0.03
*SGTR – Concurrent Spike	2.3	0.026	1.52	0.03
LB LOCA – Containment Leakage	73.2	1.46	31	0.7
LB LOCA – ECCS Recirculation	0.95	0.001	1.6	0.0027
Gas Decay Tank	n/a	0.127	n/a	0.07
Recycle Holdup Tank	0.0012	0.031	0.11	0.055
Spent Resin Tank	0.035	0.00001	0.056	0.00002
FHA	6.26	0.0838	6.8	0.17

* Results from the limiting unit

Table F-5
Comparison of Byron Control Room results

	Pre-Uprate			Post-Uprate		
	Thyroid rem	Whole Body rem	Beta Skin rem	Thyroid rem	Whole Body rem	Beta Skin rem
LOCA – Containment Leakage	18.2	2.17	23.3	18	1.8	29.7
LOCA – ECCS Recirculation	0.274	0.0000108	0.0000568	0.34	0.000014	0.00013

Table F-6
Comparison of Braidwood Control Room results

	Pre-Uprate			Post-Uprate		
	Thyroid rem	Whole Body rem	Beta Skin rem	Thyroid rem	Whole Body rem	Beta Skin rem
LOCA – Containment Leakage	16.4	2.81	32.2	18	1.8	29.5
LOCA – ECCS Recirculation	0.195	0.00000772	0.000047	0.3	0.000013	0.00013

Table F-7
Major Assumption Changes

Event	Parameter	Value Pre-uprate	Value Post- uprate
Steamline Break	RCS Coolant Concentrations	See Tables 9 and 10	See Tables 9 and 10
	Faulted SG Mass, lbm	96,000	167,000
	Primary to Secondary Leakage to Faulted SG, gpm	0.347 at hot conditions	0.5 at cold conditions
	Iodine Spike Factor, Ci/min I – 131 I – 132 I – 133 I – 134 I – 135	 87 534 200 267 192	 208 877 462 463 413
	Iodine Spike Duration, hr	2	6
	Whole Body Dose Conversion Factors	See Table 14	See Table 14
Locked Rotor	Core Inventory	See Table 8	See Table 8
	RCS Coolant Concentrations	See Table 10	See Table 10
	Primary to Secondary Leakage, gpm	1 gpm total at hot conditions	1 gpm total at cold conditions
	Thyroid Dose Conversion Factors	See Table 15	See Table 15
	Whole Body Dose Conversion Factors	See Table 14	See Table 14
Locked Rotor with Failed Open PORV	Core Inventory	n/a	See Table 8

This event is bounded by the steamline break event prior to uprate. Comparison is made to the steamline break event assumptions prior to uprate.	RCS Coolant Concentrations	See Table 10	See Table 10
	Faulted SG Mass, lbm	96,000	167,000
	Primary to Secondary Leakage to Faulted SG, gpm	0.347 at hot conditions	0.5 at cold conditions
	Failed Fuel, %	0	2
	Iodine Spike Duration, hr	2	6
	Whole Body Dose Conversion Factors	See Table 14	See Table 14
Rod Eject	Core Inventory	See Table 8	See Table 8
	RCS Coolant Concentrations	See Table 10	See Table 10
	Steam Releases	0 – 500 sec - 116,346 lbm/sec	0 – 200 sec - 3000 lbm/sec 200 – 4000 sec - 500 lbm/sec
	Fraction of Fuel Melting, %	0.25	0.375
	Thyroid Dose Conversion Factors	See Table 15	See Table 15
	Whole Body Dose Conversion Factors	See Table 14	See Table 14
Small Line Break Outside Containment	RCS Coolant Concentrations	See Tables 9 and 10	See Tables 9 and 10
	Partitioning of Iodine for Spilled Water	0.022	0.1
	Thyroid Dose Conversion Factors	See Table 15	See Table 15

ATTACHMENT 1
(continued)

	Whole Body Dose Conversion Factors	See Table 14	See Table 14
Steam generator Tube Rupture	RCS Coolant Concentrations	See Tables 9 and 10	See Tables 9 and 10
	Iodine Spike Factor, Ci/min I – 131 I – 132 I – 133 I – 134 I – 135	 343 523 768 900 696	 208 877 462 463 413
	Thyroid Dose Conversion Factors	See Table 15	See Table 15
	Whole Body Dose Conversion Factors	See Table 14	See Table 14
Large Break LOCA	Core Inventory	See Table 8	See Table 8
	Spray Removal Coefficients, hr^{-1} Elemental Iodine (injection phase)	 29.9	 20
	Unfiltered Inleakage, scfm	Byron: 78.75 Braidwood: 25	100
	Sump Water Volume, ft^3	59,920	38,979 (at beginning of ECCS recirculation) 58,506 (at beginning of spray recirculation)
	Control Room volume, ft^3	Byron: 230,837 Braidwood: 70,275	70,275 Note: A single volume is used for both stations for consistency. The original Byron analysis considered additional adjacent room volumes.
	Control Room Air Flow Rate, cfm Filtered Makeup Filtered Recirculation	 6,000 45,000	 6,000+/- 10% 39,150
	Thyroid Dose Conversion Factors	See Table 15	See Table 15
	Whole Body Dose Conversion Factors	See Table 14	See Table 14

	Spray Removal Coefficients, hr^{-1} Particulates (until DF = 50) Particulates (after DF = 50)	Not modeled	6.0 0.6
	Atmospheric Dispersion Factors, sec /m^3 Control Room – Containment Leakage 0 – 2 hr 2-8 hr 8-24 hr 24-96 hr 96-720 hr	Byron Braidwood 4.05E-3 6.24E-3 4.05E-3 624E-3 1.91E-3 3.16E-3 5.73E-4 8.4E-4 1.37E-4 1.4E-4	Byron Braidwood 6.1E-3 6.2E-3 5.3E-3 5.37E-3 2.68E-3 2.79E-3 2.0E-3 1.82E-3 1.53E-3 1.32E-3
	Atmospheric Dispersion Factors, sec /m^3 Control Room – ECCS Recirculation Leakage 0 – 2 hr 2-8 hr 8-24 hr 24-96 hr 96-720 hr	Byron Braidwood 4.05E-3 6.24E-3 4.05E-3 6.24E-3 1.91E-3 3.16E-3 5.73E-4 8.4E-4 1.37E-4 1.4E-4	Byron Braidwood 2.28E-3 2.48E-3 1.91E-3 1.87E-3 8.88E-4 8.11E-4 5.97E-4 5.04E-4 4.77E-4 3.91E-4
Gas Decay Tank Rupture	Gas Decay Tank Inventory	See Table 11	See Table 11
	Whole Body Dose Conversion Factors	See Table 14	See Table 14
Recycle Holdup Tank Rupture	Recycle Holdup Tank Inventory	See Table 12	See Table 12
	Thyroid Dose Conversion Factors	See Table 15	See Table 15
	Whole Body Dose Conversion Factors	See Table 14	See Table 14

Spent Resin Tank Rupture	Spent Resin Tank Inventory	See Table 13	See Table 13
	Thyroid Dose Conversion Factors	See Table 15	See Table 15
	Whole Body Dose Conversion Factors	See Table 14	See Table 14
Fuel Handling Accident	Core Inventory	See Table 8	See Table 8
	Time After Shutdown, hr^{-1}	100	48
	Thyroid Dose Conversion Factors	See Table 15	See Table 15
	Whole Body Dose Conversion Factors	See Table 14	See Table 14

ATTACHMENT 1
(continued)

Table F-8
Iodine and Noble Gas Inventory in Core

Nuclide	Activity (Ci) Pre-uprate	Activity (Ci) Post-Uprate
I – 131	8.80E7	9.74E7
I – 132	1.34E8	1.40E8
I – 133	1.97E8	1.97E8
I – 134	2.31E8	2.17E8
I – 135	1.79E8	1.85E8
Kr – 85m	3.95E7	2.50E7
Kr –85	9.99E5	1.02E6
Kr – 87	7.59E7	4.79E7
Kr – 88	1.08E8	6.74E7
Xe – 131m	6.68E5	1.09E6
Xe – 133m	5.16E6	6.17E6
Xe – 133	2.03E8	1.97E8
Xe – 135m	5.46E7	3.88E7
Xe – 135	5.55E7	4.00E7
Xe – 138	1.79E8	1.62E8

Table F-9
RCS Coolant Concentrations Based on 1.0 μCi/gm DE I – 131 for Iodines
and
1% Fuel Defects or Noble Gases

Nuclide	Activity (μCi/gm) Pre-uprate	Activity (μCi/gm) Post-Uprate
I – 131	0.66	0.742
I – 132	0.24	0.979
I – 133	1.06	1.350
I – 134	0.16	0.243
I – 135	0.58	0.842
Kr – 85m	2.97	1.80
Kr –85	12.46	7.11
Kr – 87	1.7	1.15
Kr – 88	5.24	3.35
Xe – 131m	-	3.31
Xe – 133m	4.39	3.65
Xe – 133	398.03	251
Xe – 135m	0.28	0.488
Xe – 135	8.92	7.72
Xe – 138	0.99	0.663

Table F-10
RCS Coolant Concentrations Based on 60.0 μCi/gm DE I – 131 for Iodines and 1% Fuel Defects or Noble Gases

Nuclide	Activity (μCi/gm) Pre-uprate	Activity (μCi/gm) Post-Uprate
I – 131	39.76	44.5
I – 132	14.31	58.7
I – 133	63.62	81.0
I – 134	9.54	14.6
I – 135	34.99	50.5
Kr – 85m	2.97	1.80
Kr –85	12.46	7.11
Kr – 87	1.7	1.15
Kr – 88	5.24	3.35
Xe – 131m	-	3.31
Xe – 133m	4.39	3.65
Xe – 133	398.03	251
Xe – 135m	0.28	0.488
Xe – 135	8.92	7.72
Xe – 138	0.99	0.663

Table F-11
Gas Decay Tank Inventory

Nuclide	Activity (Ci) Pre-uprate	Activity (Ci) Post-Uprate
Kr – 85m	5.20E+2	1.75E+2
Kr –85	6.68E+3	5.24E+3
Kr – 87	3.02E+2	4.41E+1
Kr – 88	9.02E+2	2.22E+2
Xe – 131m	4.69E+2	1.07E+3
Xe – 133m	7.63E+2	9.24E+2
Xe – 133	9.94E+4	7.51E+4
Xe – 135m	4.53E+1	7.71E+1
Xe – 135	1.53E+3	1.06E+3
Xe – 138	1.67E+2	5.98E0

Table F-12
Recycle Holdup Tank Inventory

Nuclide	Activity (Ci) Pre-uprate	Activity (Ci) Post-Uprate
I – 131	0.022	2.8
I – 132	0.008	3.7
I – 133	0.037	5.1
I – 134	0.005	0.92
I – 135	0.02	3.2
Kr – 85m	180	371
Kr –85	9100	1467
Kr – 87	69	237
Kr – 88	270	691
Xe – 131m	230	683
Xe – 133m	340	753
Xe – 133	32000	51780
Xe – 135m	2.5	101
Xe – 135	620	1593
Xe – 138	9.6	137

Table F-13
Spent Resin Tank Inventory

Nuclide	Activity (Ci) Pre-uprate	Activity (Ci) Post-Uprate
I – 131	0.92	19.6
I – 132	0.042	0.23
I – 133	0.16	2.47
I – 134	1E-3	0.027
I – 135	0.028	0.51

ATTACHMENT 1
(continued)

Table F-14
Whole Body Dose Conversion Factors

Whole Body Dose Conversion Factors	Value, pre-uprate MeV/Disintegration	Value, post-uprate rem-m^3/Ci-sec
Kr-85m	0.16	0.0371
Kr-85	0.0022	0.00051
Kr-87	0.783	0.188
Kr-88	1.934	0.466
Xe-131m	-	0.0029
Xe-133m	0.041	0.00796
Xe-133	0.045	0.00932
Xe-135m	0.432	0.0989
Xe-135	0.247	0.0574
Xe-138	1.096	0.28

Table F-15
Thyroid Dose Conversion Factors

Thyroid Dose Conversion Factors	Value, pre-uprate rem/Ci	Value, post-uprate rem/Ci
I – 131	1.48E6	1.07E6
I – 132	5.35E4	6.29E3
I – 133	4.0E5	1.81E5
I – 134	2.5E4	1.07E3
I – 135	1.24E5	3.14E4

F.7 *Present and discuss the control room dose analyses for all postulated events (include discussion of event mitigation, transport paths, and iodine and noble gas releases).*

F.7 Response

The habitability systems for Byron Station and Braidwood Station are discussed in Section 6.4 of the UFSAR. The control room is a common facility which serves both Units 1 and 2. The facility is served by two completely redundant HVAC equipment trains to meet the single-failure criterion. Two radiation monitors are provided in each control room HVAC system makeup air intake to detect high radiation. Upon detection of high radiation in the outside air intake, or upon a safety injection signal, the radiation monitoring system automatically shuts off normal outside makeup air supply to the system. The minimum outside air requirement is obtained from the turbine building makeup air intake and is routed through the makeup air filter train and fan (for removal of radioactive particulates and iodine) before being supplied to the system. The makeup air is then mixed with return air and is routed through the recirculation charcoal adsorber for the removal of radioactive iodine before being supplied to the vital areas of the control room envelope.

Two makeup air filter trains and fans are provided, each capable of handling minimum requirements of makeup air for the system. Each train is sized to process 6,000 cfm of makeup air.

The makeup air filter trains, the recirculation charcoal adsorbers, and control room shielding are designed to limit the occupational dose below levels required by General Design Criterion 19 of 10 CFR 50 Appendix A.

The entire control room envelope is designed as a low-leakage structure. During emergency operation of the control room ventilation system, such as during a radiation event, the normally open minimum outside air makeup dampers are closed. Air infiltration through dampers and through doorways during personnel ingress/egress is the only source of unfiltered air into the system.

The effective atmospheric dispersion values, χ/Q, used were calculated using the latest version of the "**A**tmospheric **R**elative **CON**centrations in Building Wakes" (ARCON96) methodology. Input data consist of hourly on-site meteorological data, release height, the building cross-sectional area which affects the dispersion of the release, the intake height, and the distance and direction from the release point to the control room air intake. Both the fresh air and turbine building air intakes were considered. The turbine building structure was conservatively assumed to not be available for the turbine building air intake case.

A continuous record for a 5-year period of on-site hourly meteorological data (i.e., January 1, 1994 through December 31, 1998) was used in the calculation. The containment surface releases considered in the calculation were conservatively treated as ground-level as they are not high enough to escape the aerodynamic effects of the plant buildings (i.e., 2.5 times containment building height). The

containment building wake effect is considered, given the release/receptor trajectories. ARCON96 calculates χ/Q values for the entire accident period (i.e., 0-2 hours, 2-8 hours, 8-24 hours, 1-4 days, and 4-30 days) using the on-site meteorological data.

An assessment of the radiological dose to control room occupants has been made for the loss-of-coolant accident (LOCA) postulated in Subsection 6.7.8 of the Power Uprate Licensing Report. The control room dose analyses for all other events are not in the licensing basis for Byron and Braidwood Stations. These analyses are included for information only.

The general parameters applicable to all control room dose analyses are provided in Table 6.7.1-3 of the Power Uprate Licensing Report. The releases of iodine and noble gases for each event are discussed in the applicable subsections of Section 6.7 of the Power Uprate Licensing Report. The control room isolation signal and isolation time are given in Table F-16. The effective atmospheric dispersion values are given in Table F-17. The resulting doses are provided in Tables F-18 and F-19 for Byron Station and Braidwood Station, respectively. The doses are within General Design Criterion 19 to 10 CFR 50, Appendix A guidelines.

ATTACHMENT 1
(continued)

Table F-16
Event Specific Control Room Dose Input Assumptions

Event	Isolation Signal	Isolation Time (minutes)
Steamline Break	Safety Injection Signal	5
Locked Rotor	High Radiation Signal	5
Locked Rotor with Failed Open PORV	High Radiation Signal	5
Rod Eject	Safety injection Signal	2.5
Small Line Break Outside Containment	High Radiation Signal	15.08
Steam generator Tube Rupture	Safety Injection Signal	U1 10.28 U2 8.06
Large Break LOCA	High Radiation Signal	0
Small Break LOCA	High Radiation Signal	0
Gas Decay Tank Rupture Recycle Holdup Tank Rupture Spent Resin Tank Rupture Fuel handling Accident	n/a emergency intake bounds fresh intake	n/a

Table F-17
Atmospheric Dispersion Factors for Control Room Dose Calculations

Event	Byron χ/Q (sec/m^3)	Braidwood χ/Q (sec/m^3)
Steamline Break, Faulted Steam Generator		
0-2 hr	1.70E-2	1.68E-2
2-8 hr	1.46E-2	1.44E-2
8-24 hr	7.24E-3	6.53E-3
24-96 hr	4.89E-3	4.47E-3
96-720 hr	3.58E-3	2.96E-3
Steamline Break, Intact Steam Generators Locked Rotor Locked Rotor with Failed PORV Steam Generator Tube Rupture		
0-0.083 hr	8.79E-3	8.71E-3
0.083-2 hr	3.98E-3	4.08E-3
2-8 hr	3.48E-3	3.43E-3
8-24 hr	1.64E-3	1.69E-3
24-96 hr	1.04E-3	9.78E-4
96-720 hr	8.96E-4	6.56E-4
Rod Eject, Containment Leakage		
0-0.0417 hr	9.82E-2	9.53E-2
0.0417-2 hr	6.10E-3	6.20E-3
2-8 hr	5.30E-3	5.37E-3
8-24 hr	2.68E-3	2.79E-3
24-96 hr	2.00E-3	1.82E-3
96-720 hr	1.53E-3	1.32E-3
Rod Eject, Steam Releases		
0-0.0417 hr	8.79E-3	8.71E-3
0.0417-2 hr	3.98E-3	4.08E-3
2-8 hr	3.48E-3	3.43E-3
8-24 hr	1.64E-3	1.69E-3
24-96 hr	1.04E-3	9.78E-4
96-720 hr	8.96E-4	6.56E-4
Small Line Break Outside Containment		
0-2 hr	2.28E-3	2.48E-3
2-8 hr	1.91E-3	1.87E-3
8-24 hr	8.88E-4	8.11E-4
24-96 hr	5.97E-4	5.04E-4
96-720 hr	4.77E-4	3.91E-4

ATTACHMENT 1
(continued)

Large Break LOCA, Containment Leakage		
0-2 hr	6.1E-3	6.2E-3
2-8 hr	5.3E-3	5.37E-3
8-24 hr	2.68E-3	2.79E-3
24-96 hr	2.0E-3	1.82E-3
96-720 hr	1.53E-3	1.32E-3
Large Break LOCA, ECCS Recirculation Gas Decay Tank Rupture Recycle Holdup Tank Rupture Spent Resin Tank Rupture Fuel Handling Accident		
0-2 hr	2.28E-3	2.48E-3
2-8 hr	1.91E-3	1.87E-3
8-24 hr	8.88E-4	8.11E-4
24-96 hr	5.97E-4	5.04E-4
96-720 hr	4.77E-4	3.91E-4
Small Break LOCA, Containment Leakage		
0-2 hr	6.10E-3	6.20E-3
2-8 hr	5.30E-3	5.37E-3
8-24 hr	2.68E-3	2.79E-3
24-96 hr	2.00E-3	1.82E-3
96-720 hr	1.53E-3	1.32E-3
Small Break LOCA, Steam Releases		
0.-2 hr	3.98E-3	4.08E-3
2-8 hr	3.48E-3	3.43E-3
8-24 hr	1.64E-3	1.69E-3
24-96 hr	1.04E-3	9.78E-4
96-720 hr	8.96E-4	6.56E-4

Table F-18
Byron Control Room Dose Results

Event	Thyroid (rem)	Whole Body (rem)	Beta Skin (rem)
Steamline Break – Preaccident Spike	14.9	0.007	0.2
Steamline Break - Concurrent Spike	28.9	0.008	0.2
Locked Rotor	2.7	0.2	2.3
Locked Rotor with Failed Open PORV	11.3	0.08	1.1
Rod Eject	25.0	0.2	2.8
Small Line Break Outside Containment - Preaccident Spike	1.2	0.004	0.08
Small Line Break Outside Containment - Concurrent Spike	0.3	0.004	0.08
Steam Generator Tube Rupture – Preaccident Spike	U1 1.39 U2 1.9	U1 0.05 U2 0.056	U1 1.21 U2 1.4
Steam Generator Tube Rupture – Concurrent Spike	U1 0.4 U2 0.52	U1 0.050 U2 0.056	U1 1.21 U2 1.4
Large Break LOCA – Containment Leakage	18	1.8	29.7
Large Break LOCA – ECCS Recirculation	0.34	0.000014	0.00013
Small Break LOCA	12.4	1.0	13.3
Gas Decay Tank Rupture	n/a	0.08	2.7
Recycle Holdup Tank Rupture	3.4	0.063	1.7
Spent Resin Tank Rupture	1.8	0.000023	0.00028
Fuel Handling Accident	2.9	0.2	5.5

Table F-19
Braidwood Control Room Dose Results

Event	Thyroid (rem)	Whole Body (rem)	Beta Skin (rem)
Steamline Break - Preaccident Spike	14.6	0.006	0.2
Steamline Break - Concurrent Spike	27.6	0.008	0.2
Locked Rotor	2.7	0.2	2.3
Locked Rotor with Failed Open PORV	11.2	0.08	1.1
Rod Eject	24.3	0.2	2.8
Small Line Break Outside Containment - Preaccident Spike	1.3	0.004	0.09
Small Line Break Outside Containment - Concurrent Spike	0.3	0.004	0.09
Steam Generator Tube Rupture – Preaccident Spike	U1 1.39 U2 1.9	U1 0.051 U2 0.057	U1 1.23 U2 1.4
Steam Generator Tube Rupture – Concurrent Spike	U1 0.4 U2 0.52	U1 0.051 U2 0.057	U1 1.23 U2 1.4
Large Break LOCA – Containment Leakage	18	1.8	29.5
Large Break LOCA – ECCS Recirculation	0.3	0.000013	0.00013
Small Break LOCA	12.3	1.0	13.6
Gas Decay Tank Rupture	n/a	0.09	2.9
Recycle Holdup Tank Rupture	3.7	0.068	1.9
Spent Resin Tank Rupture	2.0	0.000025	0.00031
Fuel Handling Accident	3.1	0.2	6.0

NRC Question Set G

G.1 *Page 6-104 indicates that the methodology used for a feedwater line break accident has been changed from the current analysis by using the reactor coolant system (RCS) thick-metal mass model of the LOFTRAN computer program. Please confirm that the this model of the LOFTRAN has been reviewed and approved by NRC staff.*

G.1 Response

The RCS thick-metal mass heat transfer model in the LOFTRAN computer program used in the analysis of the feedwater line break event for this application has not been previously reviewed and approved by the NRC staff. Therefore, as indicated on Licensing Report page 6-104, this is considered a change to the analysis methodology.

This specific model has been documented in a Supplement 1 to WCAP-7907-P-A, "LOFTRAN Code Description." A copy of Supplement 1 will be provided to the NRC Staff in response to this RAI item by January 15, 2001. Supplement 1 is a detailed description of this RCS thick-metal heat transfer model and includes additional supporting documentation of the model.

It should be noted that a less detailed RCS thick-metal mass heat transfer model in LOFTRAN has been previously approved by the NRC as documented in Westinghouse Report WCAP-8822-S1-P-A, "Mass and Energy Releases Following a Steam Line Rupture, Supplement 1 – Calculations of Steam Superheat in Mass/Energy Releases Following a Steam Line Rupture," September 1986. The thick-metal mass model is described in Section 3.1.2 of WCAP-8822-S1-P-A. The WCAP-8822-S1-P-A thick-metal mass heat transfer model is used specifically for steamline break mass and energy release calculations to conservatively maximize the heat input to the primary coolant during the cooldown. In contrast, for the Byron and Braidwood Stations feedwater line break analysis, the more detailed model described in Supplement 1 of WCAP-7907-P-A is used to credit the effects of heat removal from the primary coolant during the heatup portion of the event. This is a more realistic modeling of the transient than the typical very conservative Westinghouse methodology, which does not credit any heat transfer to RCS thick metal masses.

For this plant-specific application of the RCS thick-metal mass heat transfer model, only the following RCS components were conservatively considered in the model.

Reactor Vessel Inlet

- Reactor Vessel Shell
- Core Barrel
- Lower Head
- Lower Support Plate

Reactor Vessel Outlet
- Upper Core Plate
- Upper Support Plate

Hot Leg
- Hot Leg Piping

SG Inlet
- SG inlet plenum
- SG inlet tube sheet

SG Outlet
- SG outlet tube sheet
- SG outlet plenum

Cold Leg
- Pump Suction Leg
- Cold Leg

No other RCS components available for heat absorption were credited in this application of the RCS thick-metal mass heat transfer model.

A summary of the materials, material properties, and calculated masses of the various RCS components modeled in this application of the LOFTRAN RCS thick-metal mass heat transfer model is provided in Tables G.1-1 and G.1-2.

ATTACHMENT 1
(continued)

Table G.1-1
RCS Component Materials

RCS Lump (Section)	Lump Material	Lump Mass (lbm) (Plant Total)
REACTOR VESSEL INLET		
Reactor Vessel Shell	Carbon Steel with a Stainless Steel Clad	329408
Core Barrel	Stainless Steel	71864
Lower Head	Carbon Steel with a Stainless Steel Clad	76584
Lower Support Plate	Stainless Steel	56636
REACTOR VESSEL OUTLET		
Upper Core Plate	Stainless Steel	5724
Upper Support Plate	Stainless Steel	36236
HOT LEG	Stainless Steel	50276
STEAM GENERATOR INLET		
Inlet Plenum	Carbon Steel	21520
Tube Sheet	Inconel	24804
STEAM GENERATOR OUTLET		
Tube Sheet	Inconel	24804
Outlet Plenum	Carbon Steel	21352
COLD LEG		
Cold Leg	Stainless Steel	91788
Pump Suction Leg	Stainless Steel	52992

Table G.1-2
RCS Materials Properties

Material	Heat Capacity (BTU/lb-°F)*	Conductivity (BTU/ft-hr-°F)*	Density (lb/ft^3)
Stainless Steel	0.1299	11.02	490.0
Carbon Steel	0.1377	28.48	490.0
Inconel	0.1225	10.32	525.3

* These properties are a function of temperature. The values above correspond to a maximum initial RCS Vessel Average Termperature (Tavg) of 597.1°F.

G.2 *Page 6-137 indicates that the consequences of an inadvertent opening of a SG relief or safety valve is bounded by the main steamline break analysis discussed in Sections 6.2.4 and 6.4.5. Please provide transient departure from nucleate*

boiling (DNB) curves for the main steamline break events to confirm that there is no DNB in these events.

G.2 Response

DNBR as function of time is not specifically generated in the analysis of the steamline break event. In performing the analyses for both the inadvertent opening of a steam generator relief or safety valve and the main steamline break events, transient statepoint conditions are calculated and subsequently used in the DNB evaluation. Based on a review of these statepoint conditions, the minimum DNBR is calculated for the limiting statepoint during the transient. The main steamline break event is determined to bound the inadvertent opening of a steam generator relief of safety valve event based on a comparison of the statepoint conditions for each event. For the bounding steamline break analysis, the minimum DNBR calculated at the limiting statepoint conditions is 1.838, which is well above the DNBR limit of 1.45 for the main steamline break event with RCS pressures less than 1000 psia. The applicable DNB correlation used in determining DNBR for this event is the W-3 DNB correlation and the DNBR calculation conservatively includes the use of a 0.88 DNBR multiplier.

G.3 *Page 6-186, Item a, indicates that the SG power operated relief valves are automatically opened during a loss of normal feedwater event. Please confirm that the SG power operated relieve valves' control systems are designed to safety grade requirements or the safety analysis should not give credit to the automatic function of these relief valves.*

G.3 Response

The discussion on Page 6-186, item a, is intended to describe the normal plant response following the subject loss of normal feedwater event. It is not intended to describe the features of the plant design that are assumed in the safety analysis. The safety analysis for the loss of normal feedwater event meets all acceptance criteria; however, does not model or credit any actuation of the SG power operated relief valves (PORVs), even though the SG PORVs are safety-related.

G.4 *Page 6-193, Figure 6.2.8-1, indicated that the pressurizer level reaches 1800 cu-ft. If this curve is not accurate to reflect the results of the analysis, please provide the numerical value of the peak pressurizer level during the transient to confirm that the pressurizer does not reach a water-solid condition during a loss of main feedwater event as it is stated on page 6-190.*

G.4 Response

Figure 6.2.8-1 on page 6-193 of the Licensing Report illustrates the pressurizer water volume versus time for the loss of normal feedwater event. The water volume illustrated includes the volume of the pressurizer surge line. Therefore, for the pressurizer to reach a water-solid condition, the total pressure water volume, including the surge line, must reach a volume of 1864.44 ft^3. The peak pressurizer water volume, including the surge line, illustrated in Figure 6.2.8-1 is

1853.01 ft^3. Therefore, the pressurizer does not reach a water-solid condition for the loss of normal feedwater event.

G.5 *Please provide the results of an analysis for a loss of feedwater event regarding maximum RCS pressure (assume the main and auxiliary pressurizer sprays and pressurizer power-operated relief valves (PORVs) are inoperable, with plant initial conditions that would maximize the RCS pressure) to confirm that the peak RCS pressure is maintained below 110 percent of its design pressure.*

<u>G.5 Response</u>

The loss of feedwater event is not a RCS pressure-limiting event and, as such, it is not specifically analyzed for RCS overpressurization. The pressure transient following a loss of normal feedwater event is bounded by the more pressure limiting loss of load / turbine trip event presented in Licensing Report Section 6.2.6. The analysis for the pressure bounding loss of load/turbine trip event demonstrates that the peak RCS pressure is maintained below 110 percent of the design pressure which assumes the PORVs and pressurizer spray valves are unavailable.

G.6 *Please provide the results of an analysis for a feedwater line break accident with respect to the maximum RCS pressure to confirm that the peak RCS pressure is maintained below 110 percent of its design pressure.*

<u>G.6 Response</u>

Like the loss of normal feedwater event described in response to item G.5, the feedline break event is also not a RCS pressure-limiting event. Therefore, it too is not specifically analyzed for RCS overpressurization. The pressure transient following a feedline break event is bounded by the more limiting loss of load / turbine trip event presented in Licensing Report Section 6.2.6. The analysis for the pressure bounding loss of load/turbine trip event demonstrates that the peak RCS pressure is maintained below 110 percent of the design pressure.

G.7 *The results of the feedwater line break accident indicate that the pressurizer will reach water solid conditions. The NRC staff has generally not accepted a solid pressurizer in order to avoid the potential for all three pressurizer safety valves to be stuck open (a SBLOCA) due to liquid relief through these safety valves. Please propose necessary system modifications and provide the results of the reanalysis to confirm that the pressurizer will not reach solid conditions during this event.*

<u>G.7 Response</u>

As currently presented in Section 15.2.8 of the Byron Station and Braidwood Station UFSAR, the analysis of this American Nuclear Society (ANS) Condition IV event is not performed to demonstrate that a water solid pressurizer condition is precluded. The analysis for the feedline break event is specifically performed to demonstrate adequate heat removal capacity of the emergency auxiliary feedwater system as discussed in UFSAR Section 15.2.8, "Feedwater System

Pipe Break." Precluding pressurizer overfill in not a criterion for this event. A water solid condition in the pressurizer is not a new issue introduced by operating at uprated power conditions. No system modifications are planned due to this issue.

The initial pressurizer level assumed for the feedline break event corresponds to the nominal full power programmed level, taking into consideration pressurizer level uncertainty. This assumption is consistent with modeling of control systems in accident analyses, where uncertainties are applied to nominal operating values. TS 3.4.9 requires pressurizer level to be less than or equal to 92%. The intent of this requirement is to ensure a steam bubble exists for the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. This requirement is not intended to be used as the initial condition for accident analyses.

G.8 *Please provide the results of an analysis for reactor coolant pump (RCP) locked rotor/shaft break accident with respect to minimum transient departure from nucleate boiling ratio (DNBR) to determine the amount of fuel failure. In this analysis, a loss of offsite power should be assumed coincident with the event (see Final Safety Analysis Report, Section 15.3.3 and 15.3.5). Please assure that all fuel rods with a DNBR value below their allowable minimum DNBR should be assumed to experience DNB and become failed fuel rods. Provide transient curves which include DNBR and state the amount of failed fuel during this event.*

<u>G.8 Response</u>

The analysis of the RCP locked rotor/shaft break event presented in Licensing Report Section 6.2.12 was performed consistent with the analyses presented in UFSAR Sections 15.3.3 and 15.3.5. A concurrent loss of offsite power at the time of trip is modeled in the analysis presented in Section 6.2.12. This is evident by examination of Figure 6.2.12-1 illustrating coolant flow versus time.

The DNB analysis performed for the locked rotor/shaft break event conservatively assumes that rods with a DNBR value below the applicable limit (i.e., 1.33) fail. The percentage of rods-in-DNB determined by this specific analysis is 0.1%. To bound this value, the radiological dose analysis assumes 2.0% failed fuel for the locked rotor/shaft break event.

DNBR as function of time is not specifically generated in the analysis of locked rotor/shaft break event. In performing the analyses for this event, transient statepoint conditions were calculated and subsequently used in the DNB evaluation. Based on the limiting statepoint conditions, the percentage of rods with a minimum DNBR less than the applicable DNBR limit was determined.

G.9 *The results of the analysis for an inadvertent operation of the emergency core cooling system (ECCS) during power operation indicate that the pressurizer will reach water solid during this event. The NRC staff has generally not accepted a solid pressurizer for this accident in order to avoid the potential for all three pressurizer safety valves to be stuck open (a SBLOCA) due to liquid relief*

through these safety valves. Please propose necessary plant modifications and provide the results of your reanalysis of this event to confirm that the pressurizer will not reach water solid conditions during this event.

G.9 Response

As documented in UFSAR Section 15.5.1, "Inadvertent Operation of the Emergency Core Cooling System During Power Operation," ComEd has evaluated this event and has concluded that no modifications to the plant are necessary to prevent the pressurizer from reaching a water solid condition during this event.

Byron Station and Braidwood Station have revised Section 15.5.1 of the UFSAR crediting the pressurizer safety valves to pass water if the pressurizer reaches a water solid condition. The conclusion of this evaluation is that the transient meets its acceptance criteria even though the pressurizer may reach a water solid condition. Additionally, there are several other defenses and mitigating factors that prevent undue challenges to the pressurizer safety relief valves.

Inadvertent operation of the ECCS at power is classified as a Condition II event, a fault of moderate frequency. The acceptance criteria established for Condition II events include the following:

a. **Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values**. This criterion is met. Prior to the pressurizer reaching a water solid condition, the RCS pressure does not challenge the 2750 psia limit. Main Steam system pressure does not challenge the 110% of its design limit in this event. After the pressurizer reaches a water solid condition, the safety valves actuate to relieve the ECCS flow, maintaining RCS pressure below the 2750 psia limit.

b. **Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the DNBR limit, derived at a 95% confidence level and 95% probability.** This criterion is met.

c. **An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.** This criterion is met. The previous analysis credited operator action to manually open a pressurizer PORV during a spurious SI event. The automatic control circuitry for the PORVs is not safety related; therefore, the PORVs cannot be credited to mitigate the event. The initial licensing basis for Byron and Braidwood Stations had no acceptance criteria for pressurizer overfill. Westinghouse and the Westinghouse Owners Group (WOG) had proposed that the industry adopt the "no overfill " acceptance criterion for this event in order to resolve a Westinghouse concern that passage of water through the pressurizer safeties and the possibility that the valves may not reseat could result in a transition from a Condition II event to a Condition III event (i.e., SBLOCA). Based on the results of performance testing of safety valves in

response to NUREG-0737, Item II.D.1, Byron and Braidwood Stations have concluded that the event will not transition to a more serious plant condition.

As noted in References G.9.1 and G.9.2, (provided in Attachment G.9-1) a generic test program for pressurizer relief valves, safety valves, block valves, and associated piping systems was developed in response to NUREG-0737 Item II.D.1. This testing was performed by the Electric Power Research Institute (EPRI). Relief of subcooled water through the Crosby safety valves was part of the testing performed by EPRI. The results of the EPRI testing are summarized in EPRI Report #NP-2628-SR, "EPRI PWR Safety and Relief Valve Test Program – Safety and Relief Valve Test Report," December 1982. This report was reviewed and approved by the NRC for Byron Station and Braidwood Station in References G.9.3 and G.9.4 (provided in Attachment G.9-1) respectively.

ComEd has compared the temperatures from the EPRI subcooled water relief testing against the lowest temperature expected during a spurious SI event at Byron and Braidwood Stations, and has concluded that some valve chatter may occur; however, the resultant valve degradation will be less than the damage seen in the EPRI test. Since the EPRI tested valves were capable of closing in response to system depressurization, we have concluded that Byron and Braidwood Station valves would also be capable of closing in response to system depressurization. After use to relieve subcooled water, the safety valves may have some seat leakage through the closed valves due to the valve degradation; however, the leakage from three PSRVs would be less than the flow through one fully open PSRV. Thus, the spurious SI transient may result in a limited version of an inadvertent opening of a pressurizer safety or relief valve transient, which is also a Condition II event.

In the evaluation of an "Inadvertent Opening of a Pressurizer Safety or Relief Valve", an accidental depressurization of the RCS is postulated, potentially resulting in a release of RCS inventory into containment through a PSRV and a failed pressurizer surge tank rupture disc. No fuel damage is assumed to occur as a result of this event. As such, the radiological releases (i.e., offsite doses) resulting from this breach of the rupture disc were found to be "substantially less than that of a LOCA," but no quantifiable value is given for this dose in UFSAR Section 15,6,1. Since the leaking PSRV in the spurious SI event occurs well after the reactor has tripped, the consequences of the event are bounded by the present analysis in UFSAR Section 15.6.1.

The NRC has previously concluded in References G.9.3 and G.9.4, that the EPRI tests demonstrate that the Byron Station and Braidwood Station safety valves would be adequate to perform the required water relief function. References G.9.3 and G.9.4 also indicated the licensee must document a formal procedure to inspect the safety valves each time they discharge the loop seal water or pressurizer liquid. Both Byron Station and Braidwood Station have procedures to inspect the safety valves after loop seal water or pressurizer liquid discharge.

The radiological consequences of this event are within the limits of Condition II events. Water relief from the pressurizer PORVs may result in

overpressurization of the pressurizer relief tank (PRT), breaching the rupture disk and spilling contaminated fluid into containment. Note that there is no fuel damage as a result of this event, so the radiological release to the containment is limited to the inventory of radionuclides in the RCS coolant. The radiological releases (i.e., offsite doses) resulting from breaking the PRT rupture disk are the same for either water relief from the PORVs or the PSRVs.

Since all Condition II acceptance criteria are met, modifications and additional analyses are unnecessary.

Additional Mitigating Factors

In addition to the qualification testing of the pressurizer safety relief valves that demonstrate that these valves will perform acceptably under the challenge of water relief during the Inadvertent Operation of ECCS, there are other defenses that mitigate the challenges to these valves.

<u>Operator Action to Restore Pressurizer Power Operated Relief Valves (PORV) When Directed By Emergency Procedures</u>

In response to this event, the operators are directed by the emergency procedures to open any closed pressurizer PORV block valves. This action can be credited to occur at seven minutes following the initiating event. At this time, the operator would be expected to open one or both PORVs if the RCS pressure were above the PORV setpoint, and the PORV failed to open automatically. The pressurizer safety valves would only be subjected to water relief for less than five minutes, significantly mitigating the challenge to the valve.

<u>Operator Action to Restore Pressurizer Power Operated Relief Valves</u>

In response to this event, the operators would be expected to open any closed pressurizer PORV block valves even before the Emergency procedures directed the operator to do so. This action would be performed either immediately after the pressurizer reached a water solid condition or potentially before a reaching a water solid condition, if the operators determine a challenge to the safety valves is imminent. The operator would be expected to open one or both PORVs if the RCS pressure were above the PORV setpoint and the PORV failed to open automatically. The pressurizer safety valves would only be subjected to water relief for less than 5 minutes, significantly mitigating the challenge to the valve.

<u>Automatic operation of the Pressurizer PORVs Without Operator Action</u>

In response to this event, the pressurizer PORVs would receive an open signal prior to the safety relief valve lift setpoint. ComEd has determined that the PORV control circuitry fails to meet the IEEE criteria to be credited for automatic action in this event, but the failure to open on demand is a low probability. Further, the technical specifications allow for the PORVs to be blocked for seat leakage. This also is an infrequent configuration. The likelihood of the PORVs responding to this event provides significant defense in depth preventing the safety valves from ever being challenged with water.

Best Estimate Pressurizer Fill Time

The analysis performed for this event predicts that the pressurizer will reach a water solid condition in 3 to 4 minutes. This analysis uses several very conservative assumptions including (1) ECCS developed head provided by the charging pumps increased by 10% from pump curves, (2) initial pressurizer level at 3% greater than normal hot full power nominal level, and (3) pressurizer sprays operating. The pressurizer sprays are air operated valves. The air supply to these valves is lost due to the SI and containment isolation signal. Without these valves, the pressurizer pressure control system will not be able to supply spray as air pressure bleeds off, and the RCS pressure would be higher during the portion of the transient with the pressurizer not water solid. With the higher RCS pressure and the more realistic initial level and ECCS pump performance, the pressurizer will fill more slowly. A best estimate evaluation of this transient performed by ComEd determined that the time to fill the pressurizer is longer than 7 minutes. This added defense ensures that the operator has sufficient time to restore the PORVs and ensure their proper operation, mitigating the challenge to the safety relief valves.

Additional Mitigating Factors Conclusion

A single PORV will prevent the need for a pressurizer safety valve to actuate. Assuming a combination of automatic and/or manual PORV operation, the PORVs will mitigate the challenge to the pressurizer safety valves.

G.9 References

G.9.1 NUREG-1002, Supplement No. 1, "Safety Evaluation Report related to the operation of Braidwood Station, Units 1 and 2," Section 3.9.3.3, "Design and Installation of Pressure Relief Devices"

G.9.2 NUREG-0876, Supplement No. 5, "Safety Evaluation Report related to the operation of Byron Station, Units 1 and 2," Section 3.9.3.3, "Design and Installation of Pressure Relief Devices"

G.9.3 Letter from L. N. Olshan (NRC) to H. E. Bliss (ComEd), dated August 18, 1988; Subject: NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves for Byron Station, Units 1 and 2 (TAC Nos. 56200 and 63240) transmitting Technical Evaluation Report (TER) providing the results of the NRC's review on Byron Units 1 and 2 response to NUREG-0737, Item II.D.1

G.9.4 Letter from S. P. Sands (NRC), to T. K. Kovach (ComEd), dated May 21, 1990; Subject: NUREG-0737, Item II.D.1, Performance Testing on Relief and Safety Valves for Braidwood Station, Units 1 and 2 (TAC Nos. 64019 and 64046) transmitting Technical Evaluation Report (TER) providing the results of the NRC's review on Braidwood Units 1 and 2 response to NUREG-0737, Item II.D.1

G.10 *Please provide the results of an analysis for another case of an inadvertent operation of the ECCS during power operation, which would lead to the maximum RCS pressure which uses only safety related equipment.*

G.10 Response

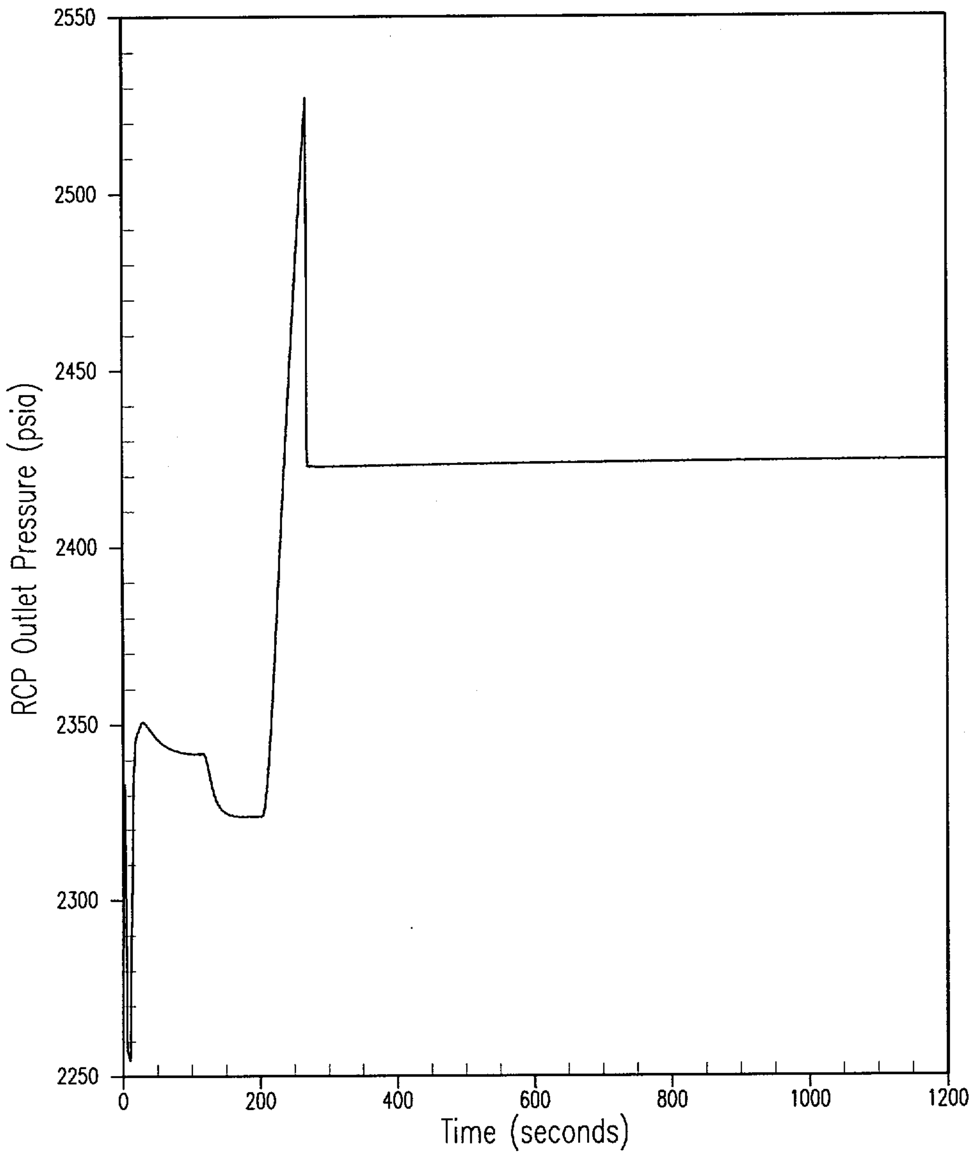
Figure G.10-1 contains the pressure trend for Section 15.6.1, "Inadvertent Operation of the Emergency Core Cooling System (ECCS) During Power Operation," from the uprated Byron and Braidwood analysis. Figure G.10-2 shows the RCS pressure response for the case which results in the maximum RCS pressure. In all cases, the event does not result in a significant RCS pressure increase until the pressurizer becomes water solid. Once the pressurizer is water solid, only the pressurizer safety valves are credited for RCS over pressure protection. The condition II acceptance criteria of 2750 psia is met.

The control system response during this event is described in UFSAR section 15.5.1.2 "Analysis Assumptions." No credit is taken for any non-safety related system in order to mitigate peak pressure. The evaluation does consider the effects of the pressurizer sprays while the pressurizer is filling (i.e., this makes the pressurizer fill faster), but the pressurizer sprays have no effect on the RCS pressure once the pressurizer is water solid.

The pressurizer sprays are modeled because it minimizes the period of time from initial event to overfill. The peak RCS pressure results of the evaluation without the pressurizer sprays are essentially identical.

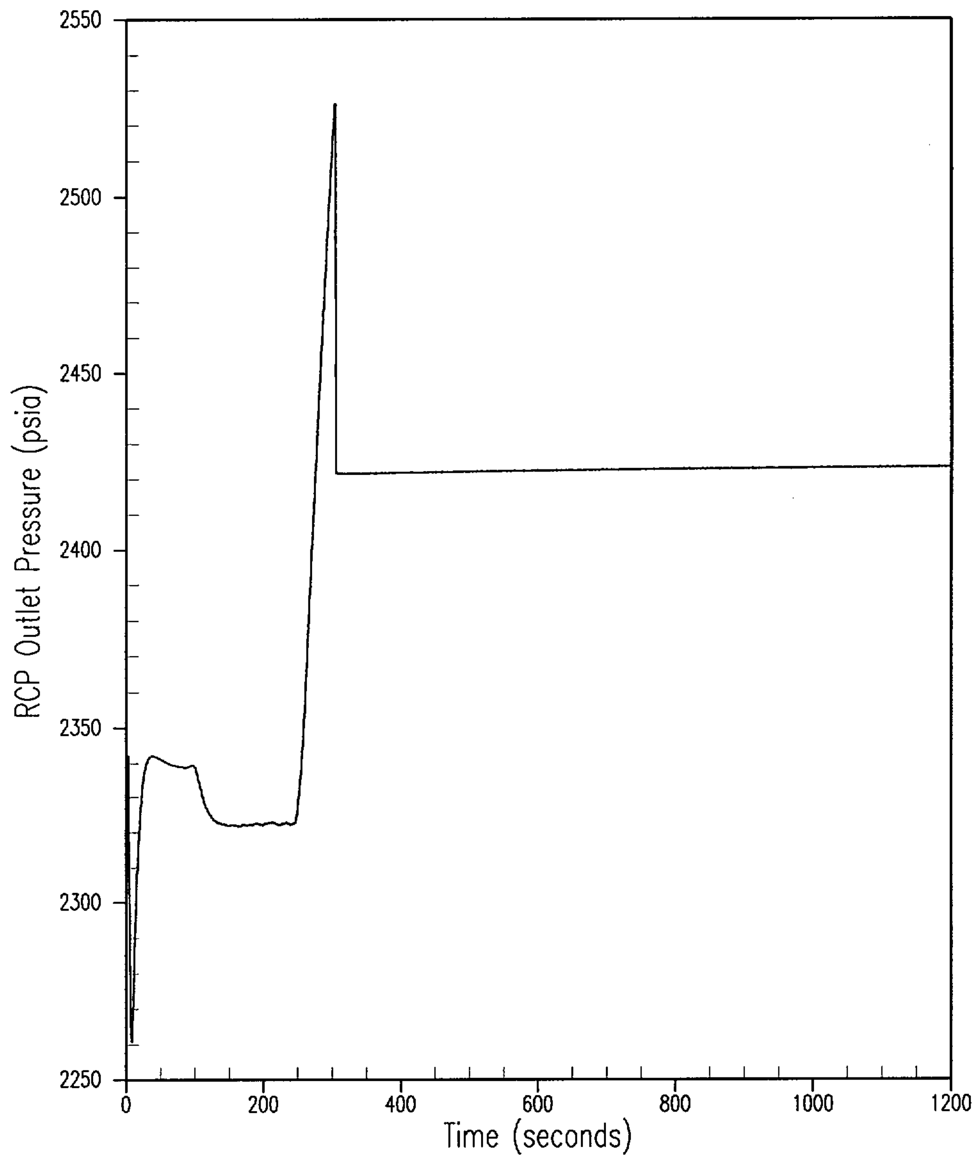
ATTACHMENT 1
(continued)

Figure G.10-1
Byron and Braidwood Units 1 (BWI Steam Generators)
RCS Cold Leg Pressure vs. Time
Inadvertent Operation of The Emergency Core Cooling System (ECCS)
During Power Operation at 3586.6 MWt



ATTACHMENT 1
(continued)

Figure G.10-2
Byron and Braidwood Units 2 (Westinghouse D5 Steam Generators)
RCS Cold Leg Pressure vs. Time
Inadvertent Operation Of The Emergency Core Cooling System (ECCS)
During Power Operation at 3586.6 MWt



G.11 *With respect to the analysis for SG tube rupture event regarding overfill, please provide transient curves for SG water levels and auxiliary feedwater flow rate to demonstrate that overfill does not occur during this event.*

G.11 Response

Transient Curves for auxiliary feedwater flow rate are provided in Figures G.11-1 and G.11-2 for Unit 1 and Unit 2, respectively.

To demonstrate that overfill does not occur during this event, transient curves for ruptured steam generator liquid volumes are provided, consistent with previous submittals for the SGTR event. Figures G.11-3 and G.11-4 show the results for Unit 1 and Unit 2, respectively. Note that Figure G.11-3 has 22 ft^3 of margin available.

Figure G.11-1
Unit 1 AF Water Flow Rate to Ruptured SG

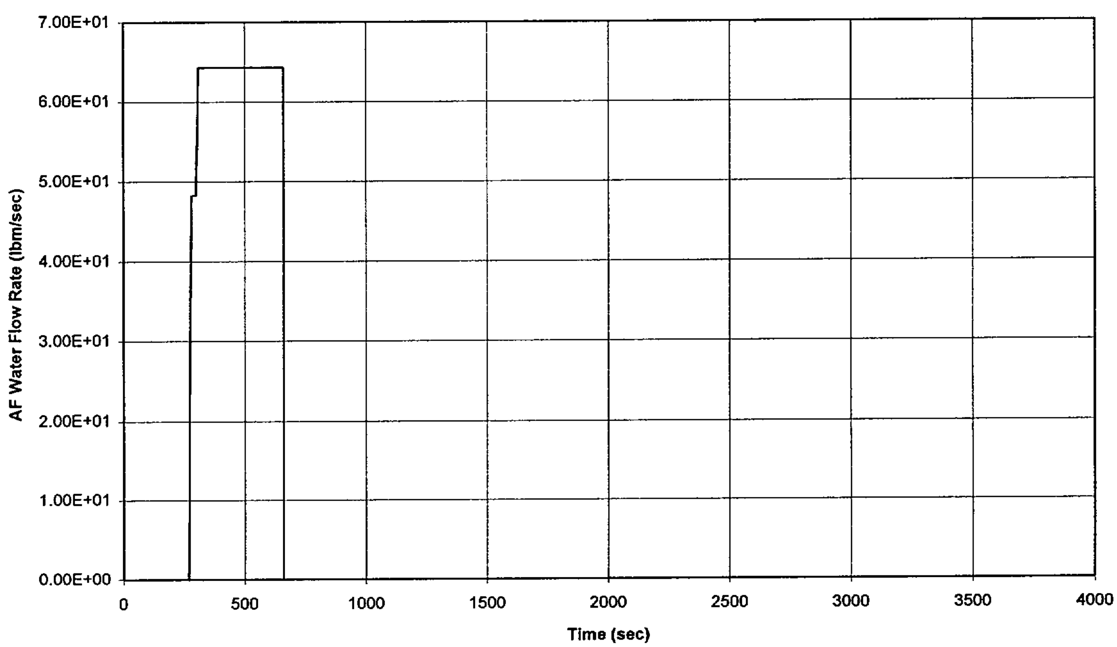


Figure G.11-2
Unit 2 AF Water Flow Rate to Ruptured SG

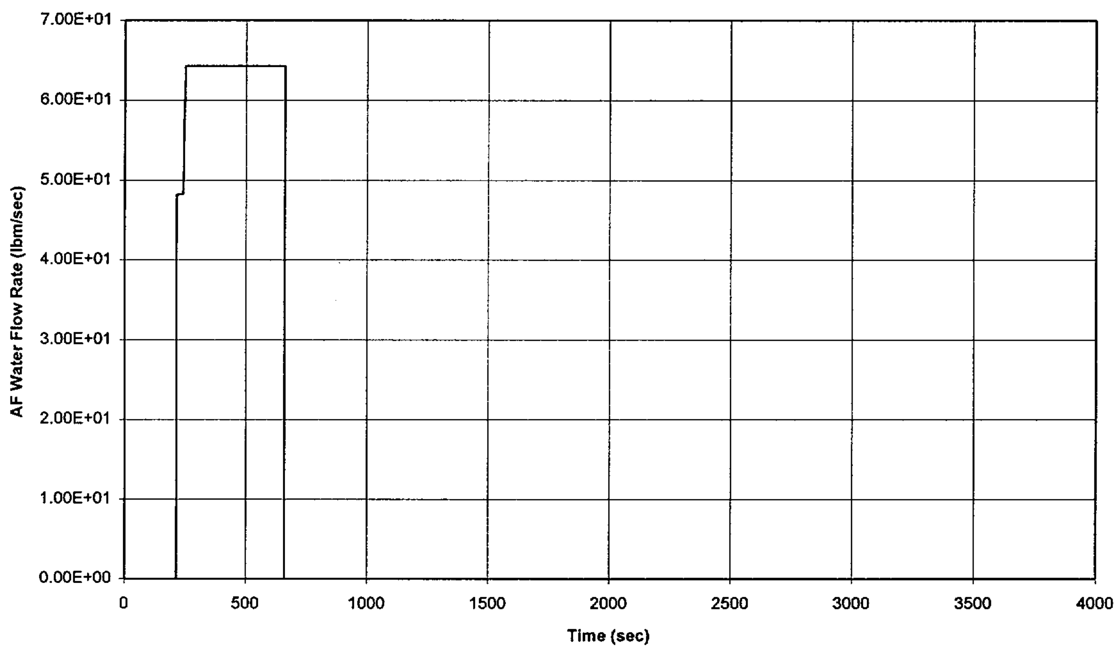


Figure G.11-3
Unit 1 Ruptured SG Liquid Volume

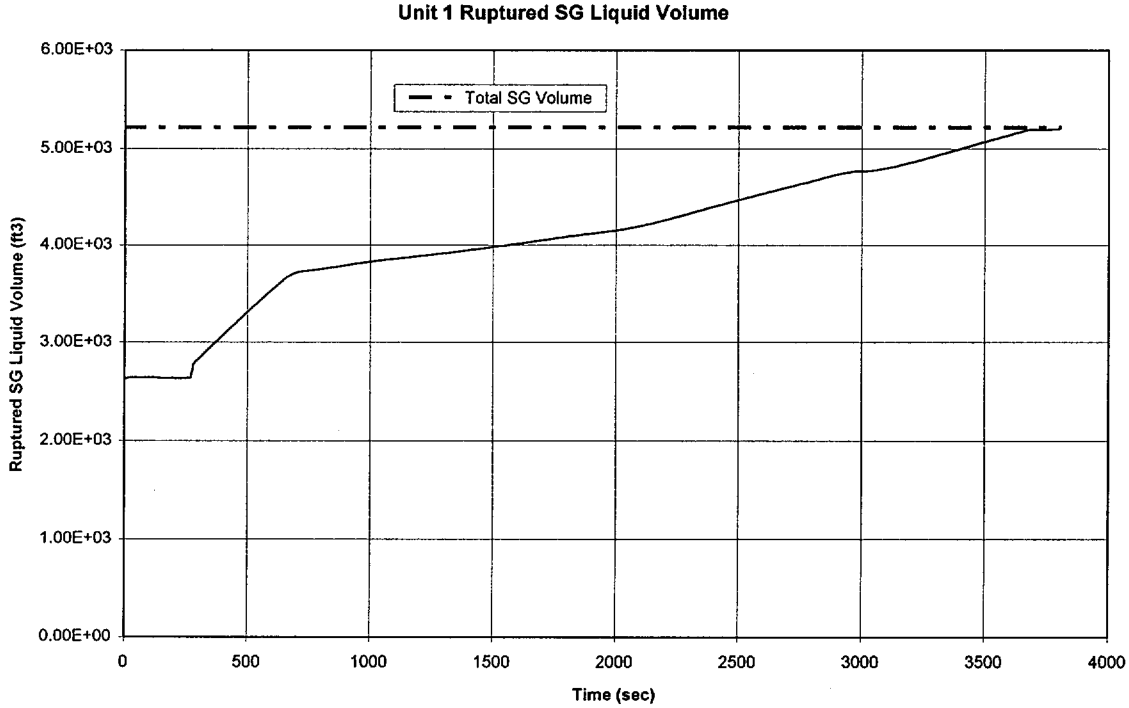
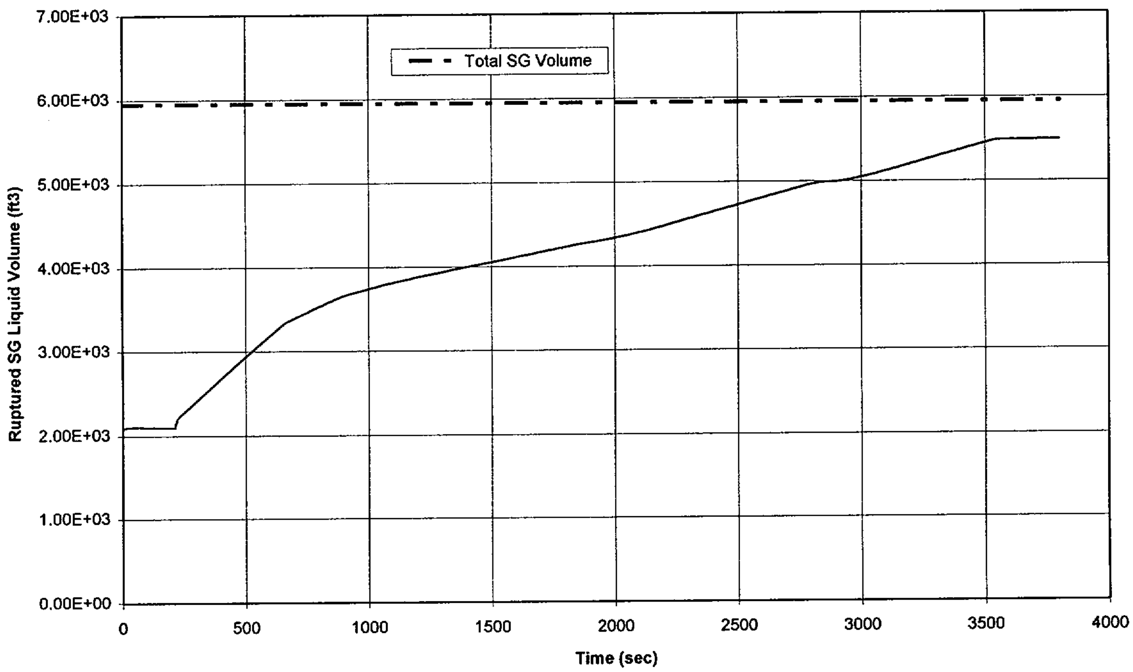


Figure G.11-4
Unit 2 Ruptured SG Liquid Volume



G.12 *Please provide the results of a thermal hydraulic analysis for a SG tube rupture based on emergency operating procedures to determine the amount of radioactive steam released to the environment and assess the radiological consequences accordingly. In this analysis, a loss of offsite power should be assumed coincident with the steam generator tube rupture (SGTR) and a most limiting single failure of a stuck open SG power operated relief valve associated with the ruptured SG. Technical specification limit of tube leakage should be assumed as an initial condition. Transient curves including DNBR, RCS pressure, SG pressures, auxiliary feedwater flow, primary to secondary leak flow rate, integrated mass leaking from the primary to secondary side, steam release from the ruptured SGs, etc., should be provided.*

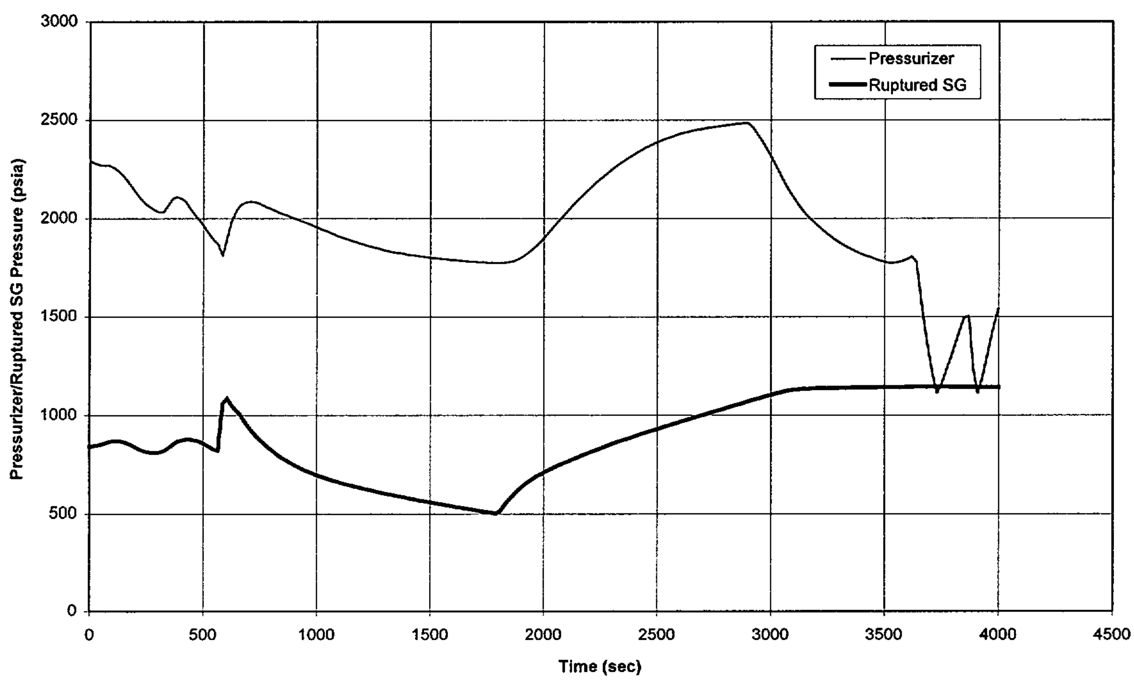
<u>G12 Response</u>

The thermal hydraulic analysis for a SG tube rupture used to determine the amount of radioactive steam released to the environment and the assessment of the radiological consequences are discussed in Section 6.7.7 of the Power Uprate Licensing Report. The operator actions assumed are based on the plant-specific emergency procedures. The operator response times assumed in the uprate analyses remain unchanged from previously approved analyses. A loss of offsite power coincident with the steam generator tube rupture (SGTR) and a most limiting single failure of a stuck open SG power operated relief valve associated with the ruptured SG are assumed. A total primary to secondary leakage of 1 gpm is assumed, which bounds the Technical Specification limit on

tube leakage. The SGTR event is bounded by the loss of flow event for DNB consideration. Section 6.2.11 of the Power Uprate Licensing Report presents the analysis for the loss of flow event and confirms that the minimum DNBR values are greater than the safety analysis limit. Consistent with approved SGTR methodology, explicit DNB analysis is not performed for the SGTR event.

Transient curves for RCS pressure, SG pressure, auxiliary feedwater flow, primary to secondary leak flow rate (ruptured tube flow), integrated mass leaking from the primary to secondary side (integrated mass from SGTR), steam release from the ruptured SG are presented in Figures G.12-1 through 5 for Unit 1. The transient curves for Unit 2 are presented in Figures G.12-6 through 10.

Figure G.12-1
Unit 1 Pressurizer/Rupture SG Pressure



ATTACHMENT 1
(continued)

Figure G.12-2
Unit 1 AF Water Flow Rate

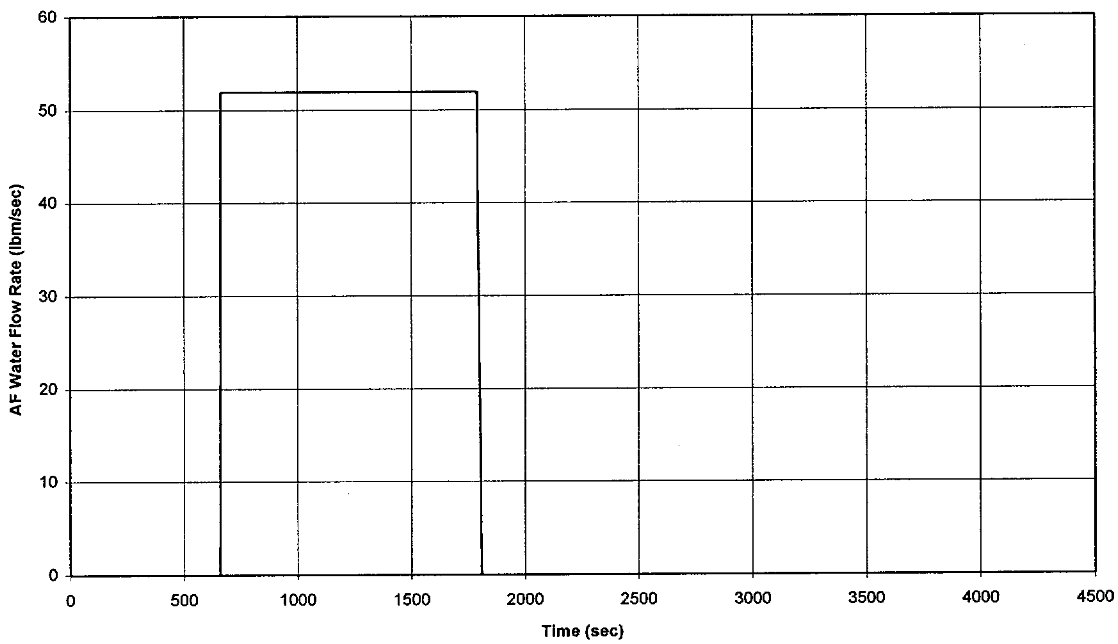


Figure G.12-3
Unit 1 Ruptured Tube Flow

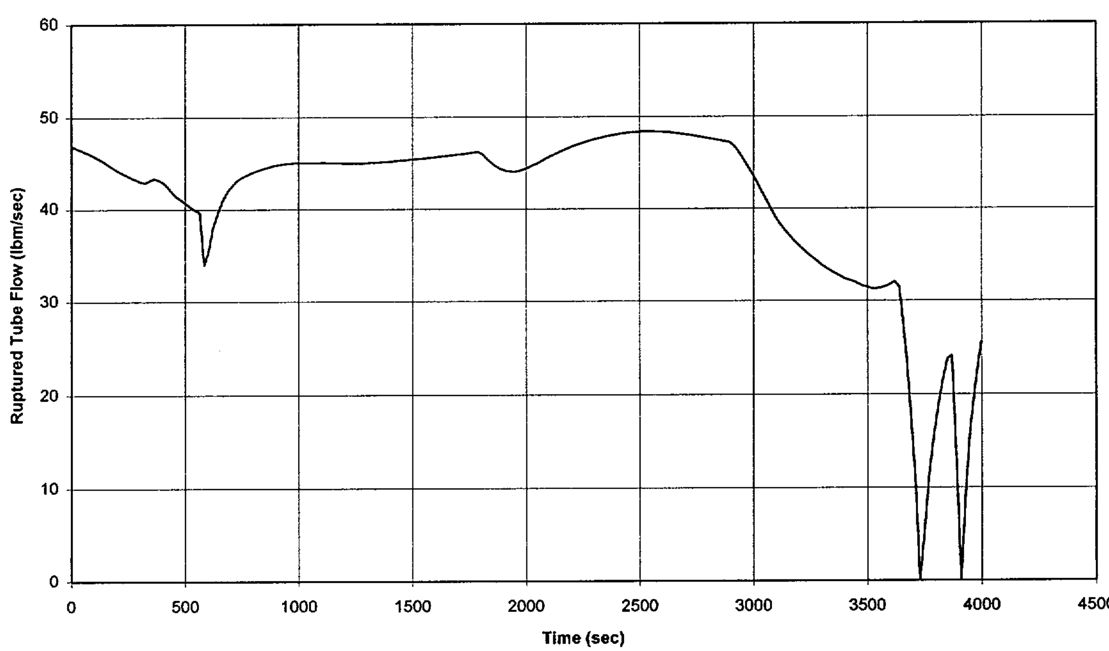


Figure G.12-4
Unit 1 Integrated Mass from SGTR

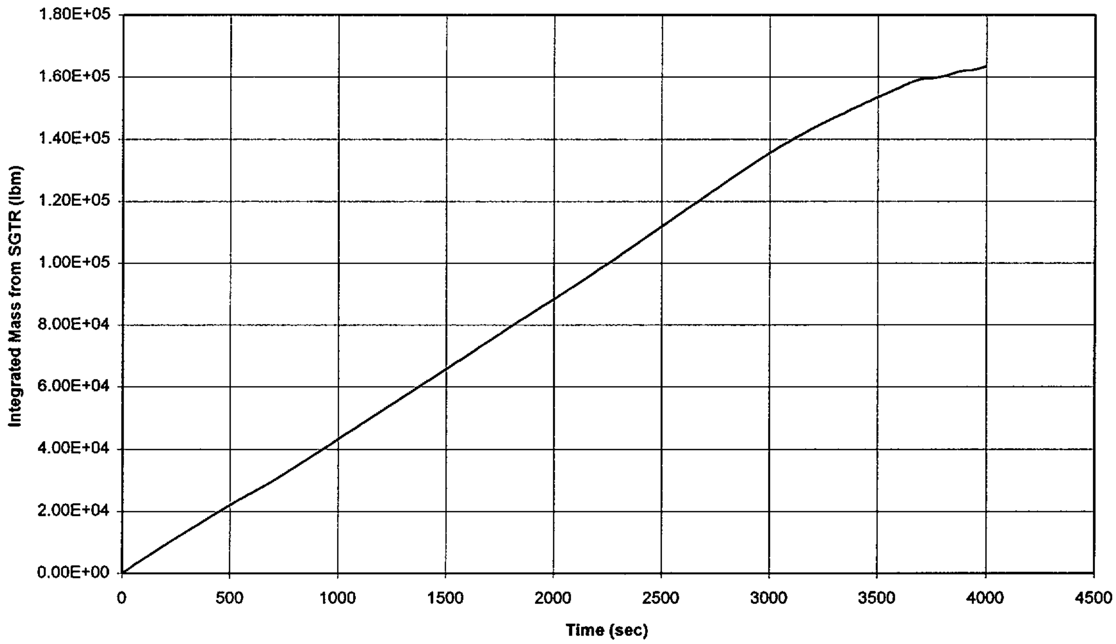
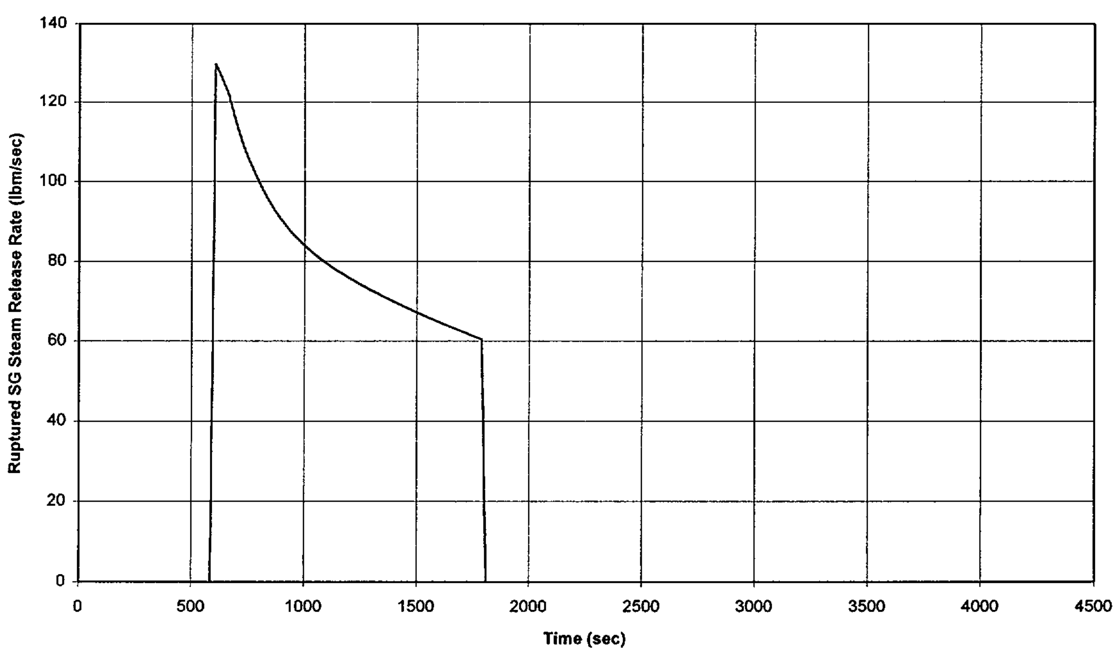


Figure G.12-5
Unit 1 Ruptured SG Steam Release Rate



ATTACHMENT 1
(continued)

Figure G.12-6
Unit 2 Pressurizer/Ruptured SG Pressure

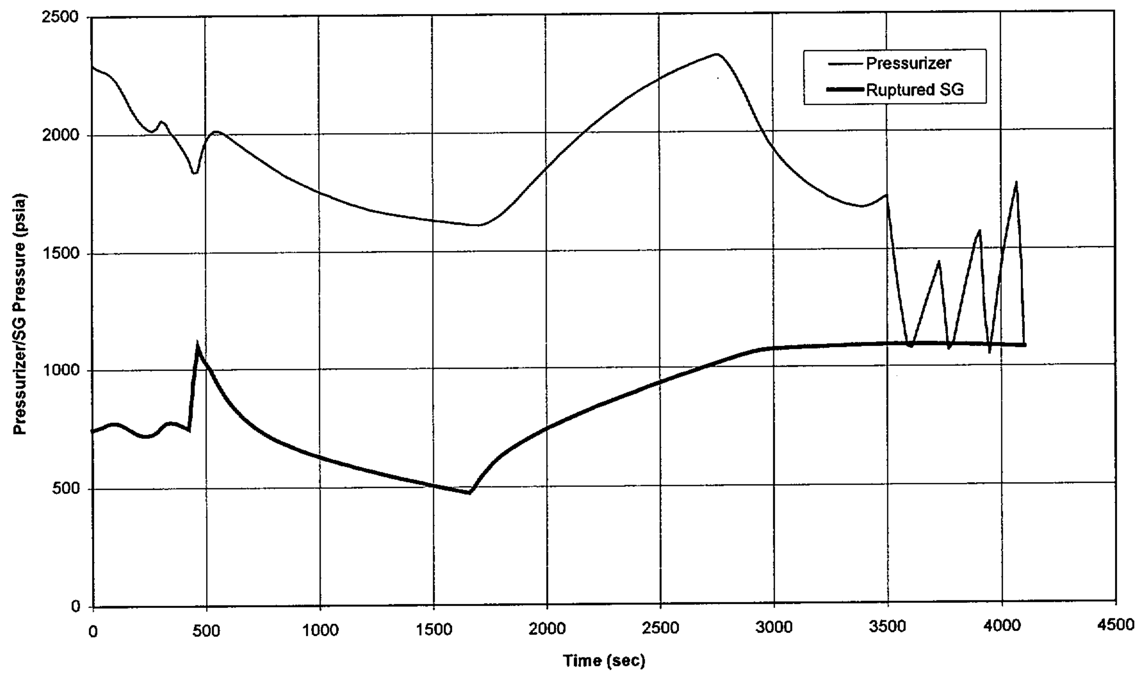


Figure G.12-7
Unit 2 AF Water Flow Rate

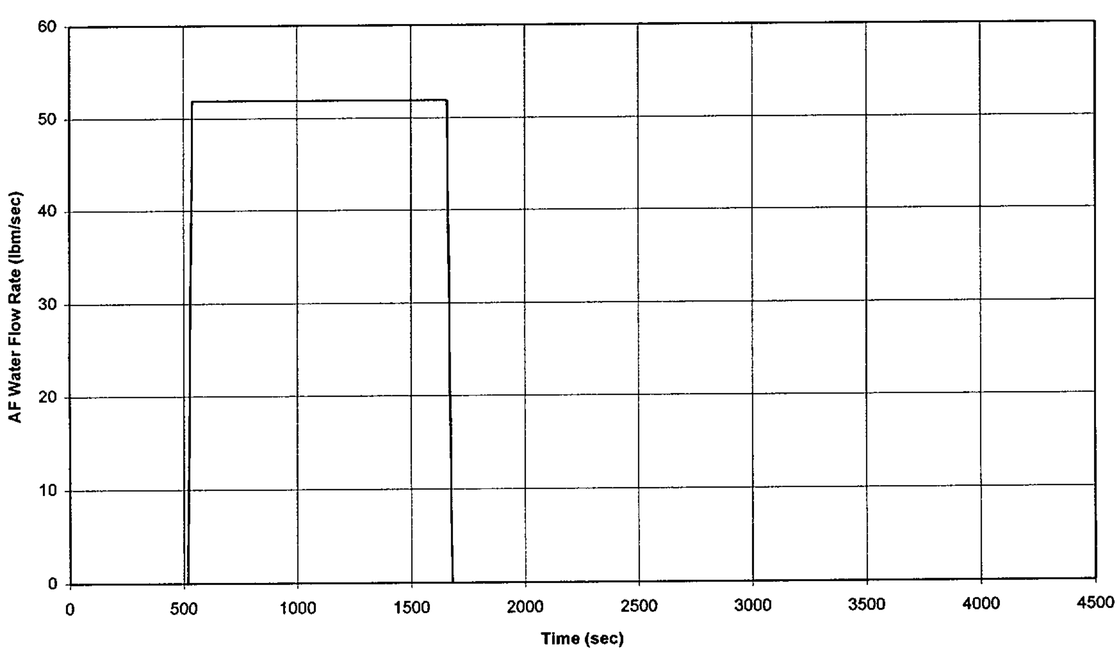


Figure G.12-8
Unit 2 Ruptured Tube Flow

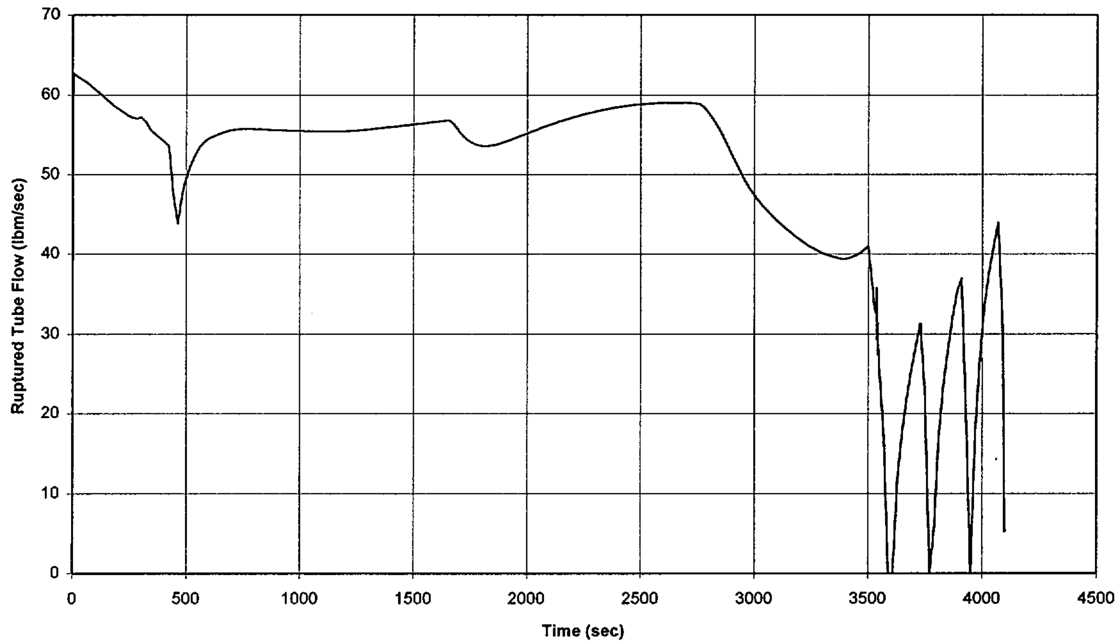
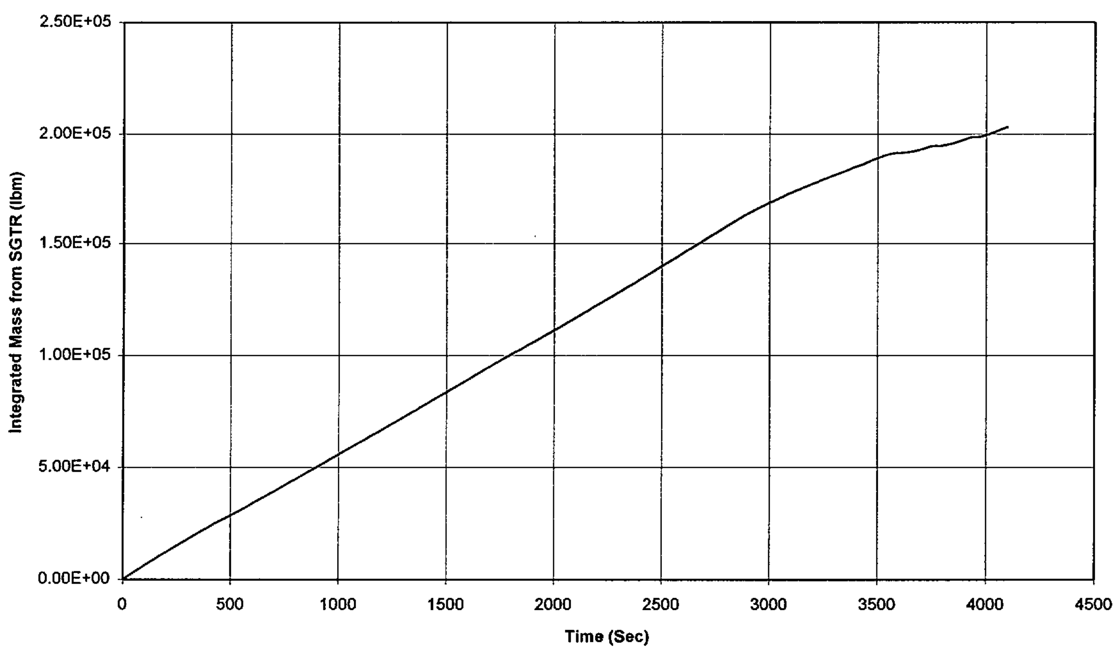
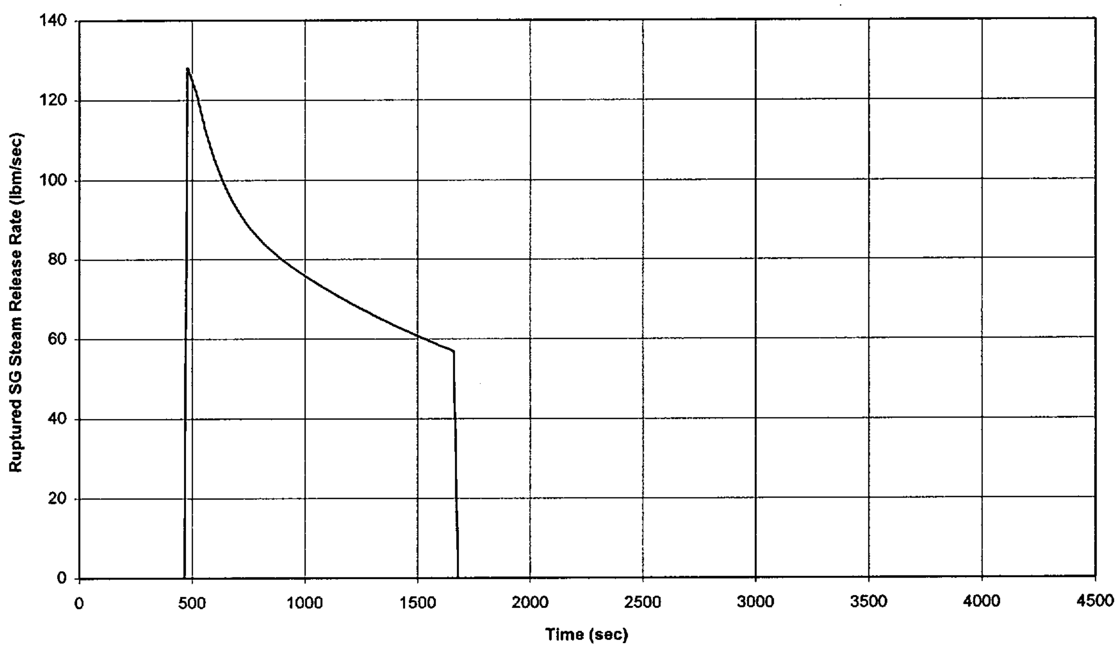


Figure G.12-9
Unit 2 Integrated Mass from SGTR



ATTACHMENT 1
(continued)

Figure G.12-10
Unit 2 Ruptured SG Steam Release Rate



NRC Verbal RAI Regarding the Power Uprate Environmental Assessment

Are the following documents applicable to Byron and Braidwood Stations, and if so, were these documents reviewed for impact due to the power uprate?

1) *Environmental Report Operating License Stage*
2) *Final Environmental Staement*
3) *Environmental Protection Plan*

Response to Environmental Assessment RAI

The following documents are applicable to the Byron Station and Braidwood Station.

- Byron Station Environmental Report Operating License Stage, issued November 30, 1978
- Braidwood Station Environmental Report Operating License Stage, issued November 30, 1978
- NUREG-0848, "Final Environmental Statement related to the operation of Byron Station, Units 1 and 2," dated April 1982
- NUREG-1026, "Final Environmental Statement related to the operation of Braidwood Station, Units 1 and 2," dated June 1984
- Byron Station Environmental Protection Plan (i.e., Appendix B to the Facility Operating License)
- Braidwood Station Environmental Protection Plan (i.e., Appendix B to the Facility Operating License)

The above documents have been reviewed to assess the potential impact on the environment due to uprated power operations. Based on this review, we have concluded that the proposed power uprate will not pose a significant adverse effect on the environment from either a radiological or non-radiological aspect.

ATTACHMENT C.1

The following documents are attached for information.

- WCAP-15391, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation"
- WCAP-15392, "Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation"
- WCAP-15390, "Evaluation of Pressurized Thermal Shock for Byron Unit 1"
- WCAP-15389, "Evaluation of Pressurized Thermal Shock for Byron Unit 2"
- WCAP-15364, "Braidwood Unit 1 Heatup and Cooldown Limit Curves for Normal Operation"
- WCAP-15373, "Braidwood Unit 2 Heatup and Cooldown Limit Curves for Normal Operation"
- WCAP-15365, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 1"
- WCAP-15381, "Evaluation of Pressurized Thermal Shock for Braidwood Unit 2"

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15391

Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation

T.J. Laubham

September 2000

Prepared by the Westinghouse Electric Company LLC
for the Commonwealth Edison Company

Approved: C H Boyd
C. H. Boyd, Manager
Engineering & Materials Technology

Approved: [signature]
D. M. Trombola, Manager
Mechanical Systems Integration

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

PREFACE

This report has been technically reviewed and verified by:

Reviewer:	Ed Terek	Ed Terek

EXECUTIVE SUMMARY

The purpose of this report is to generate new pressure-temperature limit curves for Byron Unit 1 for normal operation at 22 and 32 EFPY based of revised Uprated Fluences. The new pressure-temperature limit curves were generated using the methodology from WCAP 14040-NP-A, the 1996 ASME Boiler and Pressure Vessel Code, Section XI Appendix G, ASME Code Case N-588, ASME Code Case N-640 and WCAP-15315. Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART). The 1/4T and 3/4T ART values are summarized in Tables 4-18 and 4-19 and the limiting material is the intermediate shell forging 5P-5933. The pressure-temperature limit curves were generated for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr. The axial oriented flaw cases are limiting for all curves at each EFPY value evaluated. Hence, Code Case N-588 was not used and only the axial oriented flaw curves are presented in this report and they can be found in Figures 5-1 through 5-4.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) 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 a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{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"[1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) 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 pressure-temperature limit curves for normal operation.

NOTE: For the reactor vessel radiation surveillance program, Babcock and Wilcox Co. supplied Westinghouse with sections of SA508 Class 3 forging material used in the core region of the Byron Station Unit No. 1 reactor pressure vessel (Specifically from forging 5P-5933). Also supplied was a weldment made with weld wire heat # 442002 Linde 80 flux lot number 8873, which is identical to that used in the actual fabrication of the intermediate to lower shell girth weld of the pressure vessel).

2 PURPOSE

The Commonwealth Edison Company contracted Westinghouse to generate new pressure-temperature limit curves for 22 and 32 EFPY based on the revised fluences from the 5% Uprating. These new Pressure-Temperature Curves are to be developed utilizing the following methodologies:

- Regulatory Guide 1.99, Revision 2[1],
- ASME Code Case N-640[2],
- Elimination of the flange requirement of Appendix G to 10CFR Part 50[3] per WCAP-15315, "Reactor Vessel Head/Flange Requirements Evaluation for Operating PWR and BWR Plants"[4],
- Methodology of the 1996 ASME B&P Vessel Code, Section XI, Appendix G[6], and
- The PT Curves will be developed WITHOUT margins or instrumentation errors.

Based on the above methodologies, PT Curves will be generated consisting of the 1996 Appendix G to ASME Section XI and the K_{IC} methodologies (ASME Code Case N-640) for the limiting forging/base metal material. All curves will used the methodology to eliminate the 10 CFR Part 50 Appendix G flange requirements (from WCAP-15315). The final PT curves to be presented herein will be the most limiting set of curves.

The purpose of this report is to present the calculations and the development of the Commonwealth Edison Company Byron Unit 1 heatup and cooldown curves for 22 and 32 EFPY. This report documents the calculated adjusted reference temperature (ART) values following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

Per the request of the Commonwealth Edison Company, the surveillance weld data from the Byron Unit 1 and Unit 2 surveillance programs has been integrated. Note that Byron Unit 2 surveillance weld is identical to the surveillance weld (Heat No. 442002) at Byron Unit 1. In addition, the Braidwood Units 1 and 2 surveillance weld is identical to the nozzle to intermediate shell circumferential weld (Heat No. 442011). Per WCAP-15183[7], WCAP-15180[11] and WCAP-15368[17] all weld metal surveillance data has been determined to be credible.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[3] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix G[6], contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[2, 6] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \tag{1}$$

where, K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \tag{2}$$

where,

K_{Im}	=	stress intensity factor caused by membrane (pressure) stress
K_{It}	=	stress intensity factor caused by the thermal gradients
K_{Ic}	=	function of temperature relative 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

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \qquad (3)$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \leq \sqrt{t} \leq 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3} \times CR \times t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3} \times HU \times t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \qquad (4)$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \tag{5}$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \tag{6}$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldwon Limit Curves"[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 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 reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

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

The above procedures are needed because there is no direct control on temperature at the 1/4T location 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 ensures conservative operation of the system for the entire cooldown period.

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

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface 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 therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of 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.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G[3] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3106 psi), which is 621 psig for Byron Unit 1 reactor vessel.

This requirement was originally based on concerns about the frature margin in the closure flange region. During the boltup process, stresses in this region typically reach over 70 percent of the steady-state stress, without being at steady-state temperature. The margin of 120°F and pressure limitation of 20 percent of the hydrotest pressure were developed using the K_{IA} fracture toughness from the mid 1970's.

Improved knowledge of the fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{IC} in development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1"[2].

The discussion given in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants"[4], concluded that the integrity of the closure head/vessel flange region is not a concern for any of the operating plants using K_{IC} toughness. 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. It is therefore clear that no additional boltup requirements are necessary, and the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves contained in this report.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = Initial\ RT_{NDT} + \Delta RT_{NDT} + Margin \qquad (7)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

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

$$\Delta RT_{NDT} = CF * f^{(0.28-0.10\log f)} \qquad (8)$$

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

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \qquad (9)$$

where x inches (vessel inner radius and beltline thickness is 86.625 inches and 8.5 inches, respectively) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections for the 5% Uprating and the results are presented in SAE-REA-00-546[10]. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Table 4-1, herein, contains the calculated vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluences used to calculate the ART values for all beltline materials in the Byron Unit 1 reactor vessel. Additionally, the calculated surveillance capsule fluence values are presented in Table 4-1 and 4-2.

<u>Ratio Procedure and Temperature Adjustment:</u>

The ratio procedure, as documented in Regulatory Guide 1.99, Revision 2 Position 2.1, was used, where applicable, to adjust the measured values of ΔRT_{NDT} of the weld materials for differences in copper/nickel content. This adjustment is performed by multiplying the ΔRT_{NDT} by the ratio of the vessel chemistry factor to the surveillance material chemistry factor. The adjusted ΔRT_{NDT} values are then used to calculate the chemistry factor for the vessel materials.

From NRC Industry Meetings on November 12, 1997 and February 12, 13 of 1998, procedural guidelines were presented to adjust the ΔRT_{NDT} for temperature differences when using surveillance data from one vessel applied to another vessel. The following guidance was presented at these industry meetings:

> Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradition damage... To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}.
>
> For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{NDT,\,measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{NDT} = \Delta RT_{NDT,\,measured} + 1.0*(T_{capsule} - T_{plant}) \qquad (10)$$

The irradiation temperatures from Byron Units 1 & 2 are presented in WCAP-14824, Revision 2[12]. The average irradiation temperature from each of the four Units and operating cycles in question is 553°F. Therefore, no temperature adjustment is required.

<u>Chemistry Factor:</u>

The chemistry factor is obtained from the tables in Regulatory Guide 1.99, Revision 2 using the best estimate average copper and nickel content as reported in Tables 4-5 through 4-8. The chemistry factors were also calculated using Position 2.1 from the Regulatory Guide 1.99, Revision 2 using all available surveillance data. Per Reference 7, the surveillance weld data for Byron Unit 1 is credibile while the surveillance forging material is non-credible. In addition, Reference 7 also shows that the Table chemistry factor is conservative. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine the adjusted reference temperature of the forging material with a full margin term. Per Reference 11, the surveillance weld data for Byron Unit 2 is credibile. Position 2.1 chemistry factors are calculated in Table 4-9 and 4-10.

<u>Explanation of Margin Term:</u>

When there are "two or more credible surveillance data sets"[1] available for Byron Unit 1, Regulatory Guide 1.99 Rev. 2 (RG1.99R2) Position 2.1 states "To calculate the Margin in this case, use Equation 4; the values given there for σ_Δ may be cut in half". Equation 4 from RG1.99R2 is as follows:

$$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}\,.$$

Standard Deviation for Initial RT_{NDT} Margin Term, σ_I

If the initial RT_{NDT} values are measured values, which they are in the case of Byron Unit 1, then σ_I is equal to 0°F. On the other hand, if the initial RT_{NDT} values were not measured, then a generic value of 17°F (base metal and weld metal) would have been required to be used for σ_I .

Standard Deviation for ΔRT_{NDT} Margin Term, σ_Δ

Per RG1.99R2 Position 1.1, the values of σ_Δ are referred to as "28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}." The mean value of ΔRT_{NDT} is defined in RG1.99R2 by Equation 2 and defined herein by Equation 8.

Per RG1.99R2 Position 2.1, when there is credible surveillance data, σ_Δ is taken to be the lesser of ½ ΔRT_{NDT} or 14°F (28°F/2) for welds, or 8.5°F (17°F/2) for base metal. Where ΔRT_{NDT} again is defined herein by Equation 8.

Summary of the Margin Term

Since σ_I is taken to be zero when a heat-specific measured value of initial RT_{NDT} are available (as they are in this case), the total margin term, based on Equation 4 of RG1.99R2, will be as follows:

- Position 1.1: Lesser of ΔRT_{NDT} or 56°F for Welds
  Lesser of ΔRT_{NDT} or 34°F for Base Metal

- Position 2.1: Lesser of ΔRT_{NDT} or 28°F for Welds
  Lesser of ΔRT_{NDT} or 17°F for Base Metal

TABLE 4-1

Summary of the Peak Pressure Vessel Neutron Fluence Values at 22 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Azimuth	**Surface** (n/cm^2, E>1.0 MeV)	**¼ T** (n/cm^2, E>1.0 MeV)	**¾ T** (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 5P-5933	1.39×10^{19}	8.35×10^{18}	3.01×10^{18}
Lower Shell Forging 5P-5951	1.39×10^{19}	8.35×10^{18}	3.01×10^{18}
Nozzle Shell Forging 123J218	4.15×10^{18}	2.49×10^{18}	8.99×10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat 442002)	1.34×10^{19}	8.05×10^{18}	2.90×10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-501 (Heat 442011)	4.15×10^{18}	2.49×10^{18}	8.99×10^{17}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-2

Summary of the Peak Pressure Vessel Neutron Fluence Values at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Azimuth	**Surface** (n/cm^2, E>1.0 MeV)	**¼ T** (n/cm^2, E>1.0 MeV)	**¾ T** (n/cm^2, E>1.0 MeV)
Intermediate Shell Forging 5P-5933	2.02×10^{19}	1.21×10^{19}	4.37×10^{18}
Lower Shell Forging 5P-5951	2.02×10^{19}	1.21×10^{19}	4.37×10^{18}
Nozzle Shell Forging 123J218	6.04×10^{18}	3.63×10^{18}	1.31×10^{18}
Intermediate to Lower Shell Forging Circ. Weld Seam WF336 (Heat 442002)	1.94×10^{19}	1.16×10^{19}	4.20×10^{18}
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF501 (Heat 442011)	6.04×10^{18}	3.63×10^{18}	1.31×10^{18}

Note: All remaining vessel materials are below 1×10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-3
Calculated Integrated Neutron Exposure of the Byron Unit 1 Surveillance Capsules Tested to Date

Capsule	Fluence
U	4.04×10^{18} n/cm^2, (E > 1.0 MeV)
X	1.57×10^{19} n/cm^2, (E > 1.0 MeV)
W	2.43×10^{19} n/cm^2, (E > 1.0 MeV)

Contained in Table 4-4 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials. These measured shift values were obtained using CVGRAPH, Version 4.1[16], which is a symmetric hyperbolic tangent curve-fitting program.

TABLE 4-4
Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift(a)
Intermediate Shell Forging 5P-5933 (Tangential Orientation)	U	28.55°F
	X	9.82°F
	W	49.2°F
Intermediate Shell Forging 5P-5933 (Axial Orientation)	U	18.52°F
	X	53.03°F
	W	29.34°F
Surveillance Program Weld Metal	U	5.61°F
	X	40.11°F
	W	51.34°F
Heat Affected Zone	U	-60.2°F
	X	13.45°F
	W	15.23°F

Notes:

(a) Table 4-4 is the same as Table 4-3 from Reference 14.

Table 4-5 contains the best estimate weight percent copper and nickel for the Byron Unit 1 base materials in the beltline region. Table 4-6 contains the best estimate weight percent copper and nickel for the Byron Unit 1 surveillance weld material, while Table 4-7 presents the overall best estimate average for that heat of weld. Table 4-8 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4-9 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-11. Tables 4-9 and 4-10 provide the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-11.

TABLE 4-5

Calculation of the Best Estimate Cu and Ni Weight Percent for the Byron Unit 1 Forging Materials

Reference	**Intermediate Shell Forging 5P-5933**		**Lower Shell Forging 5P-5951**	
	Cu %	**Ni %**	**Cu %**	**Ni %**
Best Estimate Average Calculated Per Reference 14	0.04	0.74	0.04	0.64

Note: The best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-6

Calculation of the Average Cu and Ni Weight Percent for the Byron Unit 1 Surveillance Weld Material Only (Heat # 442002)

Reference	**Weight % Copper**	**Weight % Nickel**
Surveillance Weld Average Calculated Per Reference 14	0.02	0.69

Note: The best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-7

Calculation of Best Estimate Cu and Ni Weight Percent Values for the Byron Units 1 & 2 Weld Material (Using Byron 1 & 2 Chemistry Test Results)

Chemistry Type	Reference	Weight % Copper	Weight % Nickel
BEST ESTIMATE AVERAGE	14	0.04[c]	0.63[c]

NOTES:

(a) The weld material in the Byron Unit 1 surveillance program was made of the same wire and flux as the reactor vessel inter. to lower shell girth seam weld. (Weld seam WF-336, Wire Heat # 442002, Flux Type Linde 80, Flux Lot # 8873).

(b) The Byron Unit 2 surveillance weld is identical to that used in the reactor vessel core region girth seam (WF-447). The weld wire is type Linde MnMoNi (Low Cu-P), heat number 442002, with a Linde 80 type flux, lot number 8064.

(c) The best estimate chemistry values were obtained using the "average of averages" approach. In addition the best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-8[b]

Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange 124K358VA1	---	0.74	60
Vessel Flange 123J219VA1	---	0.73	10
Nozzle Shell Forging 123J218[b]	0.05	0.72	30
Intermediate Shell Forging 5P-5933	0.04	0.74	40
Lower Shell Forging 5P-5951	0.04	0.64	10
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat # 442002)	0.04	0.63	-30
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-501 (Heat # 442011)	0.03	0.67	10
Byron Unit 1 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.69	---
Byron Unit 2 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.71	---
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03	0.67, 0.71	---

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) Table duplicated from Table 4-7 of Reference 14.

TABLE 4-9[a]

Calculation of Chemistry Factors using Byron Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule f	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Intermediate Shell Forging 5P-5933 (Tangential)	U	0.404	0.748	28.55	21.36	0.560
	X	1.57	1.124	9.82	11.04	1.263
	W	2.43	1.239	49.20	60.96	1.535
Intermediate Shell Forging 5P-5933 (Axial)	U	0.404	0.748	18.52	13.85	0.560
	X	1.57	1.124	53.03	59.61	1.263
	W	2.43	1.239	29.34	36.35	1.535
				SUM:	203.17	6.716
	$CF_{5P\text{-}5933} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (203.17) \div (6.716) = \mathbf{30.3°F}$					
Byron Unit 1 Surv. Weld Material (Heat # 442002)	U	0.404	0.749	11.22 (5.61)[d]	8.40	0.561
	X	1.57	1.125	80.22 (40.11)[d]	90.25	1.266
	W	2.43	1.239	102.68 (51.34)[d]	127.22	1.535
Byron Unit 2 Surv. Weld Material (Heat # 442002)	U	0.405	0.749	16.88 (8.44)[d]	12.64	0.561
	W	1.27	1.067	57.76 (28.88)[d]	61.63	1.138
	X	2.30	1.225	108.02 (54.01)[d]	132.32	1.500
				SUM:	432.46	6.561
	$CF_{Surv.\ Weld,\ 442002} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (432.46) \div (6.561) = \mathbf{65.9°F}$					

Notes:

(a) The calculation for the Byron Unit 1 and 2 chemistry factors were taken from Reference 13 & 14

(b) FF = fluence factor = $f^{(0.28 - 0.1 * \log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (See Ref. 13 & 14).

(d) The Byron 1 & 2 surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 2.00. No temperature adjustment are required.

TABLE 4-10[a]

Calculation of Chemistry Factors using Braidwood Units 1 & 2 Surveillance Capsule Data

Material	Capsule	Capsule f	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Braidwood Unit 1 Surveillance Weld Heat 442011, WF-501	U	0.387	0.737	17.06[d]	12.57	0.543
	X	1.24	1.060	30.15[d]	31.96	1.124
	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2 Surveillance Weld Heat 442011, WF-501	U	0.400	0.746	0.0[d]	0	0.557
	X	1.23	1.058	26.3[d]	27.83	1.119
	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6.273) = 25.7°F$					

Notes:

(a) The calculation for the Braidwood Units 1 and 2 chemistry factor was taken from Reference 15.

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (See Ref. 15)

(d) The Braidwood 1 & 2 surveillance weld metal ΔRT_{NDT} values do not require a ratio factor or temperature adjustment.

TABLE 4-11

Summary of the Byron Unit 1 Reactor Vessel Beltline Material Chemistry Factors
Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	**Chemistry Factor**	
	Position 1.1	**Position 2.1**
Nozzle Shell Forging 123J218	31.0°F	---
Intermediate Shell Forging 5P-5933	26.0°F	30.3°F
Lower Shell Forging 5P-5951	26.0°F	---
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat 442002)	54.0°F	65.9°F
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-501 (Heat 442011)	41.0°F	25.7°F
Byron Unit 1 & 2 Surveillance Weld Metal	27.0°F	---
Braidwood Unit 1 & 2 Surveillance Weld Metal	41.0°F	

Contained in Tables 4-12 and 4-13 is the summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Byron Unit 1 reactor vessel beltline materials for 22 and 32 EFPY.

TABLE 4-12
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 22 EPFY Heatup/Cooldown Curves

Azimuth	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 5P-5933	8.35×10^{18}	0.949	3.01×10^{18}	0.671
Lower Shell Forging 5P-5951	8.35×10^{18}	0.949	3.01×10^{18}	0.671
Nozzle Shell Forging 123J218	2.49×10^{18}	0.623	8.99×10^{17}	0.396
Inter. to Lower Shell Forging Circ. Weld Seam WF336 (Heat 442002)	8.05×10^{18}	0.939	2.90×10^{18}	0.662
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF501 (Heat 442011)	2.49×10^{18}	0.623	8.99×10^{17}	0.396

TABLE 4-13
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 32 EPFY Heatup/Cooldown Curves

Azimuth	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate Shell Forging 5P-5933	1.21×10^{19}	1.053	4.37×10^{18}	0.770
Lower Shell Forging 5P-5951	1.21×10^{19}	1.053	4.37×10^{18}	0.770
Nozzle Shell Forging 123J218	3.63×10^{18}	0.720	1.31×10^{18}	0.473
Inter. to Lower Shell Forging Circ. Weld Seam WF336 (Heat 442002)	1.16×10^{19}	1.041	4.20×10^{18}	0.759
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF501 (Heat 442011)	3.63×10^{18}	0.720	1.31×10^{18}	0.473

Contained in Tables 4-14 through 4-17 are the calculations of the ART values used for the generation of the 22 and 32 EFPY heatup and cooldown curves.

TABLE 4-14

Calculation of the ART Values for the 1/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[(d)]	f @ 22[(a)] EFPY (x 10^{19})	¼-t f(x 10^{19})	¼-t FF	I	ΔRT_{NDT}[(c)]	σ_I	σ_Δ	M	ART[(b)]
Intermediate Shell Forging	5P-5933	0.04	0.74	26.0	1.39	0.835	0.949	40	24.7	0	12.3	24.7	89
Intermediate Shell Forging → using S/C Data[(e)]				30.3	1.39	0.835	0.949	40	28.8	0	17.0	34.0	**103**
Lower shell Forging	5P-5951	0.04	0.64	26.0	1.39	0.835	0.949	10	24.7	0	12.3	24.7	59
Inter. to Lower Shell Circ. Weld Metal	WF-336	0.04	0.63	54.0	1.34	0.805	0.939	-30	50.7	0	25.4	50.7	71
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.34	0.805	0.939	-30	61.9	0	14.0	28.0	60
Nozzle Shell Forging	123J218	0.05	0.72	31.0	0.415	0.249	0.623	30	19.3	0	9.7	19.3	69
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-501	0.03	0.67	41.0	0.415	0.249	0.623	10	25.5	0	12.8	25.5	61
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.415	0.249	0.623	10	16.0	0	8.0	16.0	42

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002). The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

(e) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine ART with a full margin term.

TABLE 4-15

Calculation of the ART Values for the 3/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 22[a] EFPY ($x 10^{19}$)	¾-t f($x 10^{19}$)	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	5P-5933	0.04	0.74	26.0	1.39	0.301	0.671	40	17.4	0	8.7	17.4	75
Intermediate Shell Forging → using S/C Data[e]				30.3	1.39	0.301	0.671	40	20.3	0	17.0	34.0	94
Lower shell Forging	5P-5951	0.04	0.64	26.0	1.39	0.301	0.671	10	17.4	0	8.7	17.4	45
Inter. to Lower Shell Circ. Weld Metal	WF-336	0.04	0.63	54.0	1.34	0.290	0.662	-30	35.7	0	17.9	35.7	41
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.34	0.290	0.662	-30	43.6	0	14.0	28.0	42
Nozzle Shell Forging	123J218	0.05	0.72	31.0	0.415	0.0899	0.396	30	12.3	0	6.1	12.3	55
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-501	0.03	0.67	41.0	0.415	0.0899	0.396	10	16.2	0	8.1	16.2	42
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.415	0.0899	0.396	10	10.2	0	5.1	10.2	30

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

(e) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine ART with a full margin term.

TABLE 4-16

Calculation of the ART Values for the 1/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF(d)	f @ 32(a) EFPY (x 10^{19})	¼-t f(x 10^{19})	¼-t FF	I	ΔRT_{NDT}(c)	σ_I	σ_Δ	M	ART(b)
Intermediate Shell Forging	5P-5933	0.04	0.74	26.0	2.02	1.21	1.053	40	27.4	0	13.7	27.4	95
Intermediate Shell Forging → using S/C Data(e)				30.3	2.02	1.21	1.053	40	31.9	0	17.0	34.0	**106**
Lower shell Forging	5P-5951	0.04	0.64	26.0	2.02	1.21	1.053	10	27.4	0	13.7	27.4	65
Inter. to Lower Shell Circ. Weld Metal	WF-336	0.04	0.63	54.0	1.94	1.16	1.041	-30	56.2	0	28.0	56.0	82
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.94	1.16	1.041	-30	68.6	0	14.0	28.0	67
Nozzle Shell Forging	123J218	0.05	0.72	31.0	0.604	0.363	0.720	30	22.3	0	11.2	22.3	75
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-501	0.03	0.67	41.0	0.604	0.363	0.720	10	29.5	0	14.8	29.5	69
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.604	0.363	0.720	10	18.5	0	9.3	18.5	47

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

(e) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine ART with a full margin term.

TABLE 4-17

Calculation of the ART Values for the 3/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[(d)]	f @ 32[(a)] EFPY (x 10^{19})	¾-t f(x 10^{19})	¾-t FF	I	ΔRT_{NDT}[(c)]	σ_I	σ_Δ	M	ART[(b)]
Intermediate Shell Forging	5P-5933	0.04	0.74	26.0	2.02	.437	.770	40	20.0	0	10.0	20.0	80
Intermediate Shell Forging → using S/C Data[(e)]				30.3	2.02	.437	.770	40	23.3	0	17.0	34.0	97
Lower shell Forging	5P-5951	0.04	0.64	26.0	2.02	.437	.770	10	20.0	0	10.0	20.0	50
Inter. to Lower Shell Circ. Weld Metal	WF-336	0.04	0.63	54.0	1.94	.420	.759	-30	41.0	0	20.5	41.0	52
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.94	.420	.759	-30	50.0	0	14.0	28.0	48
Nozzle Shell Forging	123J218	0.05	0.72	31.0	0.604	0.131	0.473	30	14.7	0	7.3	14.7	59
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-501	0.03	0.67	41.0	0.604	0.131	0.473	10	19.4	0	9.7	19.4	49
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.604	0.131	0.473	10	12.2	0	6.1	12.2	34

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

(e) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine ART with a full margin term.

The intermediate shell forging 5P-5933 is the limiting beltline material for all heatup and cooldown curves to be generated. The ART values associated with this material will be used in all sets of curves. Contained in Tables 4-18 and 4-19 is a summary of the limiting ARTs to be used in the generation of the Byron Unit 1 reactor vessel heatup and cooldown curves.

TABLE 4-18

Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4T Location for 22 EFPY

Material	22 EFPY	
	1/4T ART	**3/4T ART**
Intermediate Shell Forging 5P-5933	89	75
- Using Surveillance Data[a]	**103**[a]	**94**[a]
Lower Shell Forging 5P-5951	59	45
Circumferential Weld WF-336	71	41
- Using Credible Surveillance Data	60	42
Circumferential Weld WF-501	61	42
- Using Credible Surveillance Data form Braidwood 1 and 2	42	30
Nozzle Shell Forging 123J218	69	55

<u>NOTES:</u>

(a) These ART values were used to calculate the Heatup and cooldown curves in Figure 5-1 and 5-2.

TABLE 4-19

Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4T Location for 32 EFPY

Material	**32 EFPY**	
	1/4T ART	**3/4T ART**
Intermediate Shell Forging 5P-5933	95	80
- Using Surveillance Data	**106[(a)]**	**97[(a)]**
Lower Shell Forging 5P-5951	65	50
Circumferential Weld WF-336	82	52
- Using Credible Surveillance Data	67	48
Circumferential Weld WF-501	69	49
- Using Credible Surveillance Data form Braidwood 1 and 2	47	34
Nozzle Shell Forging 123J218	75	59

NOTES:

(a) These ART values were used to calculate the Heatup and cooldown curves in Figure 5-3 and 5-4.

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Section 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1 through 5-4 present the 22 and 32 EFPY heatup and cooldown curves (without margins for possible instrumentation errors) for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr using the 1996 Appendix G methodology[6]. Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-4. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1 and 5-3 (for the specific heatup rate being utilized). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640[2] (approved in February 1999) as follows:

$$1.5K_{Im} < K_{Ic} \qquad (11)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,

$K_{Ic} = 33.2 + 20.734\, e^{[0.02\,(T - RT_{NDT})]}$,

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 3. The pressure-temperature 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 temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3 of this report. The minimum temperatures for the inservice hydrostatic leak test for the Byron Unit 1 reactor vessel at 22 and 32 EFPY are 163°F, 166°F at 2485 psig 1996 App. G Methodology. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-4 define all of the above limits for ensuring prevention of nonductile failure for the Byron Unit 1 reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-4 are presented in Tables 5-1 through 5-4, respectively.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING

LIMITING ART VALUES AT 22 EFPY: 1/4T, 103°F

3/4T, 94°F

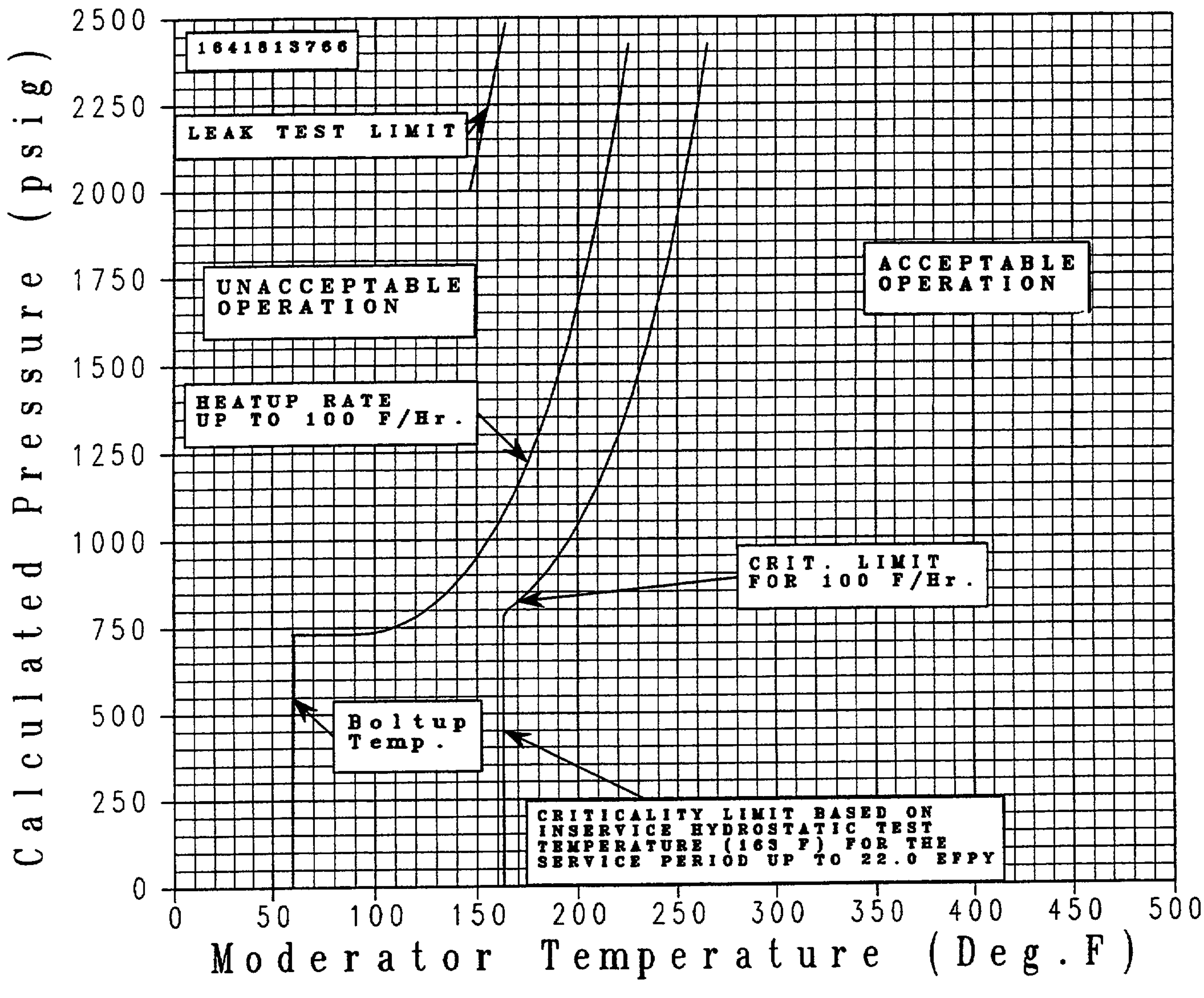


FIGURE 5-1 Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 22 EFPY using 1996 Appendix G (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING
LIMITING ART VALUES AT 22 EFPY: 1/4T, 103°F
3/4T, 94°F

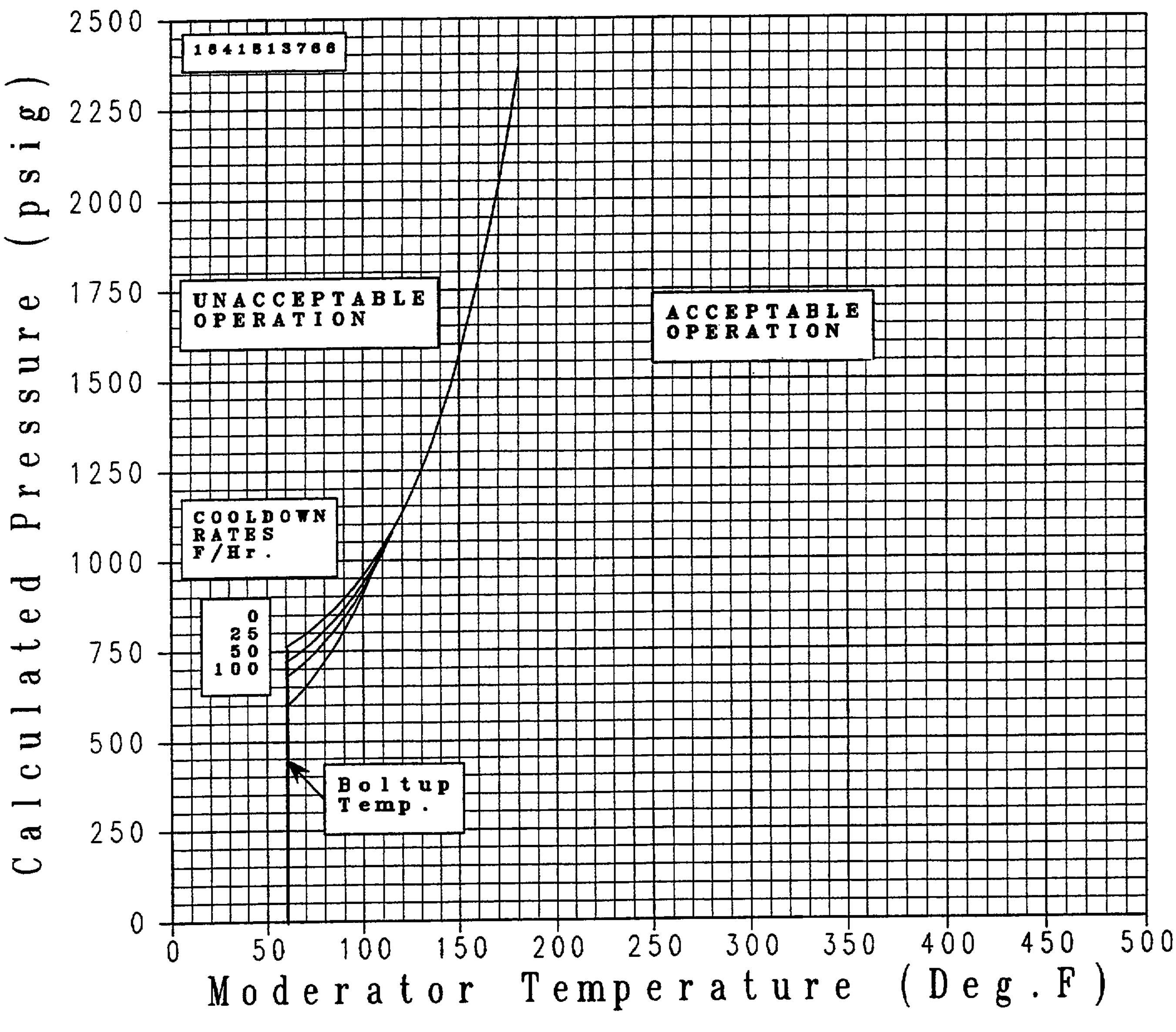


FIGURE 5-2 Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 22 EFPY using.1996 Appendix G (Without Margins of for Instrumentation Errors)

TABLE 5-1
Byron Unit 1 Heatup Data at 22 EFPY Using 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

Heatup Curves					
100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	163	0	146	2000
60	732	163	732	163	2485
65	732	163	732		
70	732	163	732		
75	732	163	732		
80	732	163	732		
85	732	163	732		
90	732	163	733		
95	733	163	737		
100	737	163	743		
105	743	163	753		
110	753	163	766		
115	766	163	782		
120	782	165	801		
125	801	170	823		
130	823	175	849		
135	849	180	879		
140	879	185	913		
145	913	190	951		
150	951	195	994		
155	994	200	1042		
160	1042	205	1095		
165	1095	210	1155		
170	1155	215	1221		
175	1221	220	1294		
180	1294	225	1376		
185	1376	230	1466		
190	1466	235	1566		
195	1566	240	1676		
200	1676	245	1798		
205	1798	250	1932		
210	1932	255	2081		
215	2081	260	2245		
220	2245	265	2425		
225	2425				

TABLE 5-2
Byron Unit 1 Cooldown Data at 22 EFPY Using 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	763	60	719	60	677	60	596
65	779	65	738	65	698	65	622
70	798	70	759	70	721	70	651
75	818	75	781	75	746	75	684
80	841	80	807	80	774	80	720
85	866	85	834	85	806	85	760
90	894	90	865	90	840	90	804
95	924	95	900	95	879	95	853
100	958	100	937	100	921	100	908
105	995	105	979	105	969	105	969
110	1037	110	1026	110	1021	110	1036
115	1082	115	1077	115	1079		
120	1133						
125	1188						
130	1250						
135	1318						
140	1393						
145	1476						
150	1568						
155	1669						
160	1781						
165	1905						
170	2042						
175	2194						
180	2361						

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING

LIMITING ART VALUES AT 32 EFPY: 1/4T, 106°F

3/4T, 97°F

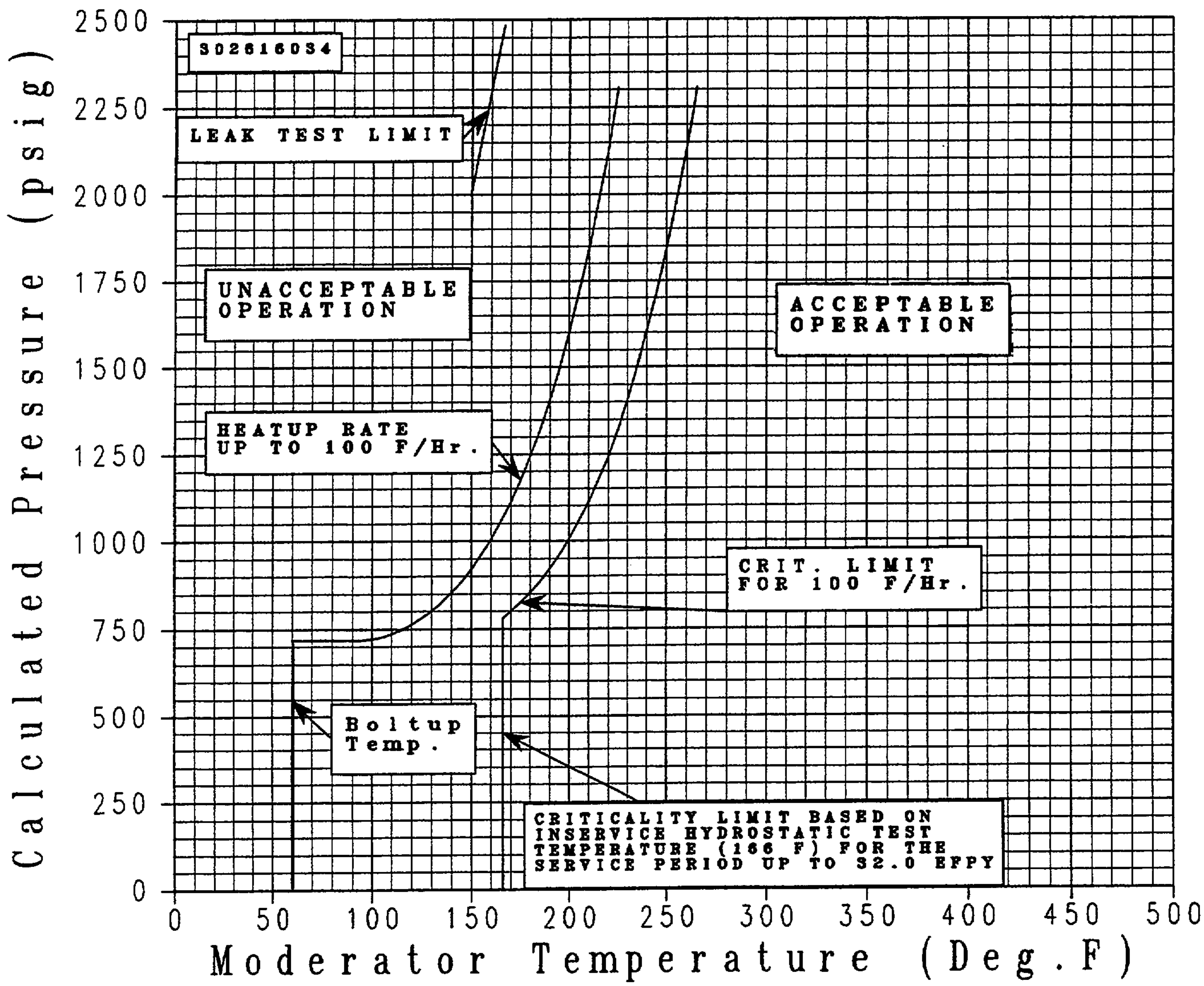


FIGURE 5-3 Byron Unit 1 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 32 EFPY Using 1996 Appendix G (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: INTERMEDIATE SHELL FORGING

LIMITING ART VALUES AT 32 EFPY: 1/4T, 106°F
3/4T, 97°F

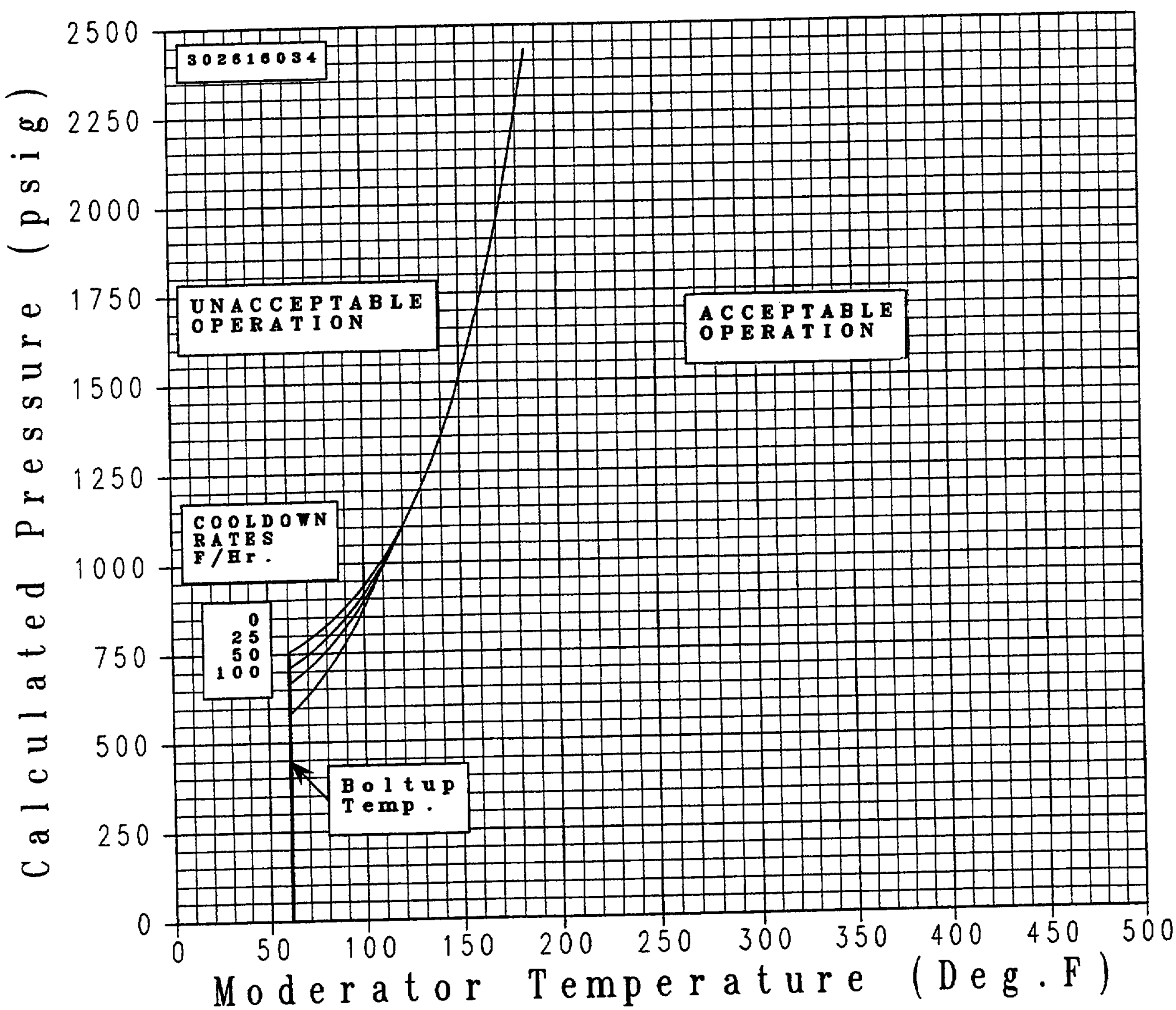


FIGURE 5-4 Byron Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY Using 1996 Appendix G (Without Margins of for Instrumentation Errors)

TABLE 5-3
Byron Unit 1 Heatup Data at 32 EFPY Using 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

Heatup Curves					
100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	166	0	149	2000
60	720	166	720	166	2485
65	720	166	720		
70	720	166	720		
75	720	166	720		
80	720	166	720		
85	720	166	720		
90	720	166	720		
95	720	166	723		
100	723	166	729		
105	729	166	737		
110	737	166	749		
115	749	166	764		
120	764	166	781		
125	781	170	802		
130	802	175	826		
135	826	180	854		
140	854	185	886		
145	886	190	921		
150	921	195	962		
155	962	200	1007		
160	1007	205	1057		
165	1057	210	1113		
170	1113	215	1175		
175	1175	220	1244		
180	1244	225	1321		
185	1321	230	1406		
190	1406	235	1499		
195	1499	240	1603		
200	1603	245	1718		
205	1718	250	1844		
210	1844	255	1984		
215	1984	260	2138		
220	2138	265	2308		
225	2308				

TABLE 5-4
Byron Unit 1 Cooldown Data at 32 EFPY Using 1996 App. G Methodology
(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	753	60	709	60	665	60	581
65	769	65	726	65	685	65	606
70	787	70	746	70	706	70	633
75	806	75	767	75	730	75	663
80	827	80	791	80	757	80	697
85	851	85	817	85	786	85	735
90	877	90	846	90	819	90	777
95	906	95	879	95	855	95	823
100	937	100	914	100	895	100	874
105	973	105	954	105	940	105	931
110	1011	110	997	110	989	110	995
115	1054	115	1045	115	1043		
120	1102	120	1099				
125	1154						
130	1212						
135	1276						
140	1347						
145	1425						
150	1512						
155	1607						
160	1713						
165	1829						
170	1958						
175	2101						
180	2258						
185	2433						

6 REFERENCES

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

2. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

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

4. WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants", W. Bamford, et.al., October 1999.

5. ASME Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels", Section XI, Division 1, Approved December 12, 1997.

6. 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.

7. WCAP-15183, "Commonwealth Edison Company Byron Unit 1 Surveillance Program Credibility Evaluation", T.J. Laubham, June 1999.

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

9. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

10. SAE-REA-00-546, "Reactor Vessel Neutron Exposure Projections for the Byron/Braidwood Uprating", J.D. Perock, January 12, 2000.

11. WCAP-15180, "Commonwealth Edison Company Byron Unit 2 Surveillance Program Credibility Evaluation", T.J. Laubham, June 1999.

12. WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance Weld Metal Integration For Byron and Braidwood", T. J. Laubham, et al., November 1997. Ref. Errata letter CAE-97-233, CCE-97-316, "Transmittal of Updated Tables to WCAP-14824 Rev. 2"

13. WCAP-15178, "Byron Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, June 1999.

14 WCAP-15124, "Byron Unit 1 Heatup and Cooldown for Normal Operation", T. J. Laubham, November 1998.

15 WCAP-15373, "Braidwood Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, March 2000.

16 CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1996.

17 WCAP-15368, "Commonwealth Edison Company Braidwood Unit 2 Surveillance Program Credibility Evaluation", T.J. Laubham, March 2000.

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15392

Byron Unit 2 Heatup and Cooldown Limit Curves for Normal Operation

T.J. Laubham

Septmeber 2000

Prepared by the Westinghouse Electric Company LLC
for the Commonwealth Edison Company

Approved: C. H. Boyd
C. H. Boyd, Manager
Engineering & Materials Technology

Approved: D. M. Trombola
D. M. Trombola, Manager
Mechanical Systems Integration

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

PREFACE

This report has been technically reviewed and verified by:

Reviewer: Ed Terek

EXECUTIVE SUMMARY

The purpose of this report is to generate new pressure-temperature limit curves for Byron Unit 2 for normal operation at 22 and 32 EFPY based of revised Uprated Fluences. The new pressure-temperature limit curves were generated using the methodology from WCAP 14040-NP-A, the 1996 ASME Boiler and Pressure Vessel Code, Section XI Appendix G, ASME Code Case N-588, ASME Code Case N-640 and WCAP-15315. Regulatory Guide 1.99, Revision 2 is used for the calculation of Adjusted Reference Temperature (ART) values. The 1/4T and 3/4T values are summarized in Tables 4-18 and 19 and were calculated using the circumferential weld WF-447, Heat 442002 (The limiting material for circumferentially oriented flaws, Code Case N-588) and nozzle shell forging 4P-6107 (The limiting material for axial flaws). The pressure-temperature limit curves were generated for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr. The axial oriented flaw cases are limiting for all curves at each EFPY value evaluated. Hence, only the axial oriented flaw curves are presented in this report and they can be found in Figures 5-1 through 5-4.

1 INTRODUCTION

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) 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 a margin. The unirradiated RT_{NDT} is designated as the higher of either the drop weight nil-ductility transition temperature (NDTT) or the temperature at which the material exhibits at least 50 ft-lb of impact energy and 35-mil lateral expansion (normal to the major working direction) minus 60°F.

RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{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"[1]. Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) 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 pressure-temperature limit curves for normal operation.

NOTE: For the reactor vessel radiation surveillance program, Babcock and Wilcox Co. supplied Westinghouse with sections of SA508 Class 3 forging material used in the core region of the Byron Station Unit No. 2 reactor pressure vessel (Specifically from forging 49D330-1/49C298-1). Also supplied was a weldment made with weld wire heat # 442002 Linde 80 flux, lot number 8064, which is identical to that used in the actual fabrication of the intermediate to lower shell girth weld of the pressure vessel).

2 PURPOSE

The Commonwealth Edison Company contracted Westinghouse to generate new pressure-temperature limit curves for 22 and 32 EFPY based on the revised fluences from the 5% Uprating. These new Pressure-Temperature Curves are to be developed utilizing the following methodologies:

- Regulatory Guide 1.99, Revision 2[1],
- ASME Code Case N-640[2],
- Elimination of the flange requirement of Appendix G to 10CFR Part 50[3] per WCAP-15315, "Reactor Vessel Head/Flange Requirements Evaluation for Operating PWR and BWR Plants"[4],
- ASME Code Case N-588[5] (where applicable),
- Methodology of the 1996 ASME B&P Vessel Code, Section XI, Appendix G[6], and
- The PT Curves will be developed WITHOUT margins or instrumentation errors.

Based on the above methodologies, two sets of PT Curves will be generated. Set one will consist of the circumferential flaw methodology (ASME Code Case N-588) in combination with 1996 Appendix G to ASME Section XI and the K_{IC} methodology (ASME Code Case N-640) for the limiting circumferential weld material. Set two will consist of the 1996 Appendix G to ASME Section XI and the K_{IC} methodology (ASME Code Case N-640) for the limiting forging/base metal material. Both sets of curves will used the methodology to eliminate the 10 CFR Part 50 Appendix G flange requirements (from WCAP-15315). The final PT curves to be presented herein will be the most limiting set of curves. If the situation arises where portions of each set of curves are limiting, then composite curves will be generated that are based on the most limiting data (i.e. Circ. Flaw or Axial Flaw Case).

The purpose of this report is to present the calculations and the development of the Commonwealth Edison Company Byron Unit 2 heatup and cooldown curves for 22 and 32 EFPY. This report documents the calculated adjusted reference temperature (ART) values following the methods of Regulatory Guide 1.99, Revision 2[1], for all the beltline materials and the development of the heatup and cooldown pressure-temperature limit curves for normal operation.

Per the request of the Commonwealth Edison Company, the surveillance weld data from the Byron Unit 1 and Unit 2 surveillance programs has been integrated. Note that Byron Unit 1 surveillance weld is identical to the surveillance weld (Heat No. 442002) at Byron Unit 2. In addition, the Braidwood Units 1 and 2 surveillance weld is identical to the nozzle to intermediate shell circumferential weld (Heat No. 442011). Per WCAP-15183[7], WCAP-15180[11] and WCAP-15368[17] all the surveillance data has been determined to be credible.

3 CRITERIA FOR ALLOWABLE PRESSURE-TEMPERATURE RELATIONSHIPS

3.1 Overall Approach

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements"[3] specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. The ASME Boiler and Pressure Vessel Code forms the basis for these requirements. Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components", Appendix G[6], contains the conservative methods of analysis.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in Code Case N-640, "Alternative Reference Fracture Toughness for Development of PT Limit Curves for Section XI"[2, 6] of the ASME Appendix G to Section XI. The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \qquad (1)$$

where, K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class1, SA-508-1, SA-508-2, SA-508-3 steel.

3.2 Methodology for Pressure-Temperature Limit Curve Development

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \qquad (2)$$

where,

K_{Im}	=	stress intensity factor caused by membrane (pressure) stress
K_{It}	=	stress intensity factor caused by the thermal gradients
K_{Ic}	=	function of temperature relative 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

For membrane tension, the corresponding K_I for the postulated defect is:

$$K_{Im} = M_m \times (pR_i / t) \tag{3}$$

where, M_m for an inside surface flaw is given by:

M_m = 1.85 for $\sqrt{t} < 2$,
M_m = $0.926\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
M_m = 3.21 for $\sqrt{t} > 3.464$

Similarly, M_m for an outside surface flaw is given by:

M_m = 1.77 for $\sqrt{t} < 2$,
M_m = $0.893\sqrt{t}$ for $2 \le \sqrt{t} \le 3.464$,
M_m = 3.09 for $\sqrt{t} > 3.464$

and p = internal pressure, Ri = vessel inner radius, and t = vessel wall thickness.

For bending stress, the corresponding K_I for the postulated defect is:

K_{Ib} = M_b * Maximum Stress, where M_b is two-thirds of M_m

The maximum K_I produced by radial thermal gradient for the postulated inside surface defect of G-2120 is $K_{It} = 0.953 \times 10^{-3}$ x CR x $t^{2.5}$, where CR is the cooldown rate in °F/hr., or for a postulated outside surface defect, $K_{It} = 0.753 \times 10^{-3}$ x HU x $t^{2.5}$, where HU is the heatup rate in °F/hr.

The through-wall temperature difference associated with the maximum thermal K_I can be determined from Fig. G-2214-1. The temperature at any radial distance from the vessel surface can be determined from Fig. G-2214-2 for the maximum thermal K_I.

(a) The maximum thermal K_I relationship and the temperature relationship in Fig. G-2214-1 are applicable only for the conditions given in G-2214.3(a)(1) and (2).

(b) Alternatively, the K_I for radial thermal gradient can be calculated for any thermal stress distribution and at any specified time during cooldown for a ¼-thickness inside surface defect using the relationship:

$$K_{It} = (1.0359C_0 + 0.6322C_1 + 0.4753C_2 + 0.3855C_3) * \sqrt{\pi a} \tag{4}$$

or similarly, K_{IT} during heatup for a ¼-thickness outside surface defect using the relationship:

$$K_{It} = (1.043C_0 + 0.630C_1 + 0.481C_2 + 0.401C_3) * \sqrt{\pi a} \tag{5}$$

where the coefficients C_0, C_1, C_2 and C_3 are determined from the thermal stress distribution at any specified time during the heatup or cooldown using the form:

$$\sigma(x) = C_0 + C_1(x/a) + C_2(x/a)^2 + C_3(x/a)^3 \tag{6}$$

and x is a variable that represents the radial distance from the appropriate (i.e., inside or outside) surface to any point on the crack front and a is the maximum crack depth.

Note, that equations 3, 4 and 5 were implemented in the OPERLIM computer code, which is the program used to generate the pressure-temperature (P-T) limit curves. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. Therefore, the P-T curve methodology is unchanged from that described in WCAP-14040, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldwon Limit Curves"[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 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 reference flaw of Appendix G to the ASME Code is assumed to exist at the inside of the vessel wall. During cooldown, the controlling location of the flaw is always at the inside of the wall because the thermal gradients produce tensile stresses at the inside, which increase with increasing cooldown rates. Allowable pressure-temperature relations are generated for both steady-state and finite cooldown rate situations. From these relations, composite limit curves are constructed for each cooldown rate of interest.

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

The above procedures are needed because there is no direct control on temperature at the 1/4T location 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 ensures conservative operation of the system for the entire cooldown period.

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

The second portion of the heatup analysis concerns the calculation of the pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface 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 therefore tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rates, each heatup rate must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced by constructing a composite curve based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of 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.

Code Case N-588; Circumferential Welds:

In 1997, ASME Section XI, Appendix G was revised to add methodology for the use of circumferential flaws when considering circumferential welds in developing pressure-temperature limit curves. This change was also implemented in a separate Code Case, N-588.

The earlier ASME Section XI, Appendix G approach mandated the postulation of an axial flaw in circumferential welds for the purposes of calculating pressure-temperature limits. Postulating the Appendix G reference flaw in a circumferential weld is physically unrealistic because the length of the reference flaw is 1.5 times the vessel thickness and is much longer than the width of the vessel girth welds.

In addition, historical experience, with repair weld indications found during pre-service inspection and data taken from destructive examination of actual vessel welds, confirms that any flaws are small, laminar in nature and are not oriented transverse to the weld bead orientation. Because of this, any defects potentially introduced during fabrication process (and not detected during subsequent non-destructive examinations) should only be oriented along the direction of the weld fabrication. Thus, for circumferential welds, any postulated defect should be in the circumferential orientation.

The revision to Appendix G now eliminates additional conservatism in the assumed flaw orientation for circumferential welds. The following revisions were made to ASME Section XI, Appendix G:

G-2214.1 Membrane Tension...

The K_I corresponding to membrane tension for the postulated circumferential defect of -2120 is

$$K_{Im} = M_m \times (pR_i / t)$$

where, M_m for an inside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Similarly, M_m for an outside surface flaw is given by:

$$M_m = 0.89 \text{ for } \sqrt{t} < 2,$$
$$M_m = 0.443\sqrt{t} \text{ for } 2 \leq \sqrt{t} \leq 3.464,$$
$$M_m = 1.53 \text{ for } \sqrt{t} > 3.464$$

Note again, that the only change relative to the OPERLIM computer code was the addition of the constants for M_m in a circ. weld limited condition. No other changes were made to the OPERLIM computer code with regard to P-T calculation methodology. As stated previously, the P-T curve methodology is unchanged from that described in WCAP-14040[8] Section 2.6 (equations 2.6.2-4 and 2.6.3-1) with the exceptions just described above.

3.3 Closure Head/Vessel Flange Requirements

10 CFR Part 50, Appendix G[3] addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3106 psi), which is 621 psig for Byron Unit 2 reactor vessel.

This requirement was originally based on concerns about the frature margin in the closure flange region. During the boltup process, stresses in this region typically reach over 70 percent of the steady-state stress, without being at steady-state temperature. The margin of 120°F and pressure limitation of 20 percent of the hydrotest pressure were developed using the K_{IA} fracture toughness from the mid 1970's.

Improved knowledge of the fracture toughness and other issues which affect the integrity of the reactor vessel have led to the recent change to allow the use of K_{IC} in development of pressure-temperature curves, as contained in Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1"[2].

The discussion given in WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Operating PWR and BWR Plants"[4], concluded that the integrity of the closure head/vessel flange region is not a concern for any of the operating plants using K_{IC} toughness. 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. It is therefore clear that no additional boltup requirements are necessary, and the requirement of 10 CFR Part 50, Appendix G, can be eliminated from the Pressure-Temperature Curves contained in this report.

4 CALCULATION OF ADJUSTED REFERENCE TEMPERATURE

From Regulatory Guide 1.99, Revision 2, the adjusted reference temperature (ART) for each material in the beltline region is given by the following expression:

$$ART = Initial\, RT_{NDT} + \Delta RT_{NDT} + Margin \qquad (7)$$

Initial RT_{NDT} is the reference temperature for the unirradiated material as defined in paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code[9]. If measured values of initial RT_{NDT} for the material in question are not available, generic mean values for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class.

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

$$\Delta RT_{NDT} = CF * f^{(0.28-0.10\log f)} \qquad (8)$$

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

$$f_{(depth x)} = f_{surface} * e^{(-0.24x)} \qquad (9)$$

where x inches (vessel inner radius and beltline thickness is 86.625 inches and 8.5 inches, respectively) is the depth into the vessel wall measured from the vessel clad/base metal interface. The resultant fluence is then placed in Equation 8 to calculate the ΔRT_{NDT} at the specific depth.

The Westinghouse Radiation Engineering and Analysis group evaluated the vessel fluence projections for the 5% Uprating and the results are presented in SAE-REA-00-546[10]. The evaluation used the ENDF/B-VI scattering cross-section data set. This is consistent with the methods presented in WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves"[8]. Tables 4-1 and 4-2, herein, contain the calculated vessel surface fluence values along with the Regulatory Guide 1.99, Revision 2, 1/4T and 3/4T calculated fluences used to calculate the ART values for all beltline materials in the Byron Unit 2 reactor vessel. Additionally, the calculated surveillance capsule fluence values are presented in Table 4-3.

Ratio Procedure and Temperature Adjustment:

The ratio procedure, as documented in Regulatory Guide 1.99, Revision 2 Position 2.1, was used, where applicable, to adjust the measured values of ΔRT_{NDT} of the weld materials for differences in copper/nickel content. This adjustment is performed by multiplying the ΔRT_{NDT} by the ratio of the vessel chemistry factor to the surveillance material chemistry factor. The adjusted ΔRT_{NDT} values are then used to calculate the chemistry factor for the vessel materials.

From NRC Industry Meetings on November 12, 1997 and February 12, 13 of 1998, procedural guidelines were presented to adjust the ΔRT_{NDT} for temperature differences when using surveillance data from one vessel applied to another vessel. The following guidance was presented at these industry meetings:

> Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradition damage... To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}.
>
> For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{NDT,\ measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{NDT} = \Delta RT_{NDT,\ measured} + 1.0*(T_{capsule} - T_{plant}) \quad (10)$$

The irradiation temperatures from Byron Units 1 & 2 are presented in WCAP-14824, Revision 2[12]. The average irradiation temperature from each of the four Units and operating cycles in question is 553°F. Therefore, no temperature adjustment is required.

Chemistry Factor:

The chemistry factor is obtained from the tables in Regulatory Guide 1.99, Revision 2 using the best estimate average copper and nickel content as reported in Tables 4-5 through 4-8. The chemistry factors were also calculated using Position 2.1 from the Regulatory Guide 1.99, Revision 2 using all available surveillance data. Per Reference 11, the surveillance weld data and the lower shell forging data for Byron Unit 2 is credible. In addition, per Reference 7, the surveillance weld data for Byron Unit 1 is credible. Position 2.1 chemistry factors are calculated in Tables 4-9 and 4-10. Reference17, the Braidwood 1 & 2 surveillance weld data is credible.

Explanation of Margin Term:

When there are "two or more credible surveillance data sets"[1] available for Byron Unit 2, Regulatory Guide 1.99 Rev. 2 (RG1.99R2) Position 2.1 states "To calculate the Margin in this case, use Equation 4; the values given there for σ_Δ may be cut in half". Equation 4 from RG1.99R2 is as follows:

$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$.

Standard Deviation for Initial RT_{NDT} Margin Term, σ_I

If the initial RT_{NDT} values are measured values, which they are in the case of Byron Unit 2, then σ_I is equal to 0°F. On the other hand, if the initial RT_{NDT} values were not measured, then a generic value of 17°F (base metal and weld metal) would have been required to be used for σ_I.

Standard Deviation for ΔRT_{NDT} Margin Term, σ_Δ

Per RG1.99R2 Position 1.1, the values of σ_Δ are referred to as "28°F for welds and 17°F for base metal, except that σ_Δ need not exceed 0.50 times the mean value of ΔRT_{NDT}." The mean value of ΔRT_{NDT} is defined in RG1.99R2 by Equation 2 and defined herein by Equation 8.

Per RG1.99R2 Position 2.1, when there is credible surveillance data, σ_Δ is taken to be the lesser of ½ ΔRT_{NDT} or 14°F (28°F/2) for welds, or 8.5°F (17°F/2) for base metal. Where ΔRT_{NDT} again is defined herein by Equation 8.

Summary of the Margin Term

Since σ_I is taken to be zero when a heat-specific measured value of initial RT_{NDT} are available (as they are in this case), the total margin term, based on Equation 4 of RG1.99R2, will be as follows:

- Position 1.1: Lesser of ΔRT_{NDT} or 56°F for Welds
  Lesser of ΔRT_{NDT} or 34°F for Base Metal
- Position 2.1: Lesser of ΔRT_{NDT} or 28°F for Welds
  Lesser of ΔRT_{NDT} or 17°F for Base Metal

TABLE 4-1

Summary of the Peak Pressure Vessel Neutron Fluence Values at 22 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Inter. Shell Forging 49D329-1/49C297-1	1.41 x 10^{19}	8.47 x 10^{18}	3.05 x 10^{18}
Lower Shell Forging 49D330-1/49C298-1	1.41 x 10^{19}	8.47x 10^{18}	3.05 x 10^{18}
Nozzle Shell Forging 4P-6107	3.58 x 10^{18}	2.15 x 10^{18}	7.75 x 10^{17}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	1.39 x 10^{19}	8.35 x 10^{18}	3.01 x 10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	3.58 x 10^{18}	2.15 x 10^{18}	7.75 x 10^{17}

Note: All remaining vessel materials are below 1 x 10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-2

Summary of the Peak Pressure Vessel Neutron Fluence Values at 32 EFPY used for the Calculation of ART Values (n/cm^2, E > 1.0 MeV)

Material	Surface (n/cm^2, E>1.0 MeV)	¼ T (n/cm^2, E>1.0 MeV)	¾ T (n/cm^2, E>1.0 MeV)
Inter. Shell Forging 49D329-1/49C297-1	2.06 x 10^{19}	1.24 x 10^{19}	4.46 x 10^{18}
Lower Shell Forging 49D330-1/49C298-1	2.06 x 10^{19}	1.24 x 10^{19}	4.46 x 10^{18}
Nozzle Shell Forging 4P-6107	5.22 x 10^{18}	3.13 x 10^{18}	1.13 x 10^{18}
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	2.03 x 10^{19}	1.22 x 10^{19}	4.40 x 10^{18}
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	5.22 x 10^{18}	3.13 x 10^{18}	1.13 x 10^{18}

Note: All remaining vessel materials are below 1 x 10^{17} n/cm^2, E > 1.0 MeV

TABLE 4-3
Calculated Integrated Neutron Exposure of the Byron Unit 2
Surveillance Capsules Tested to Date

Capsule	Fluence
U	4.05×10^{18} n/cm^2, (E > 1.0 MeV)
W	1.27×10^{19} n/cm^2, (E > 1.0 meV)
X	2.30×10^{19} n/cm^2, (E > 1.0 MeV)

Contained in Table 4-4 is a summary of the Measured 30 ft-lb transition temperature shifts of the beltline materials. These measured shift values were obtained using CVGRAPH, Version 4.1[16], which is a symmetric hyperbolic tangent curve-fitting program.

TABLE 4-4
Measured 30 ft-lb Transition Temperature Shifts of the Beltline Materials Contained in the Surveillance Program

Material	Capsule	Measured 30 ft-lb Transition Temperature Shift[(a)]
Intermediate Shell Forging	U	-3.8°F
49D330-1/49C298-1	W	3.65°F
(Tangential Orientation)	X	15.75°F
Intermediate Shell Forging	U	19.76°F
49D330-1/49C298-1	W	31.88°F
(Axial Orientation)	X	38.91°F
Surveillance Program	U	8.44°F
Weld Metal	W	28.88°F
(Heat # 442002)	X	54.01°F
Heat Affected Zone	U	6.74°F
	W	30.44°F
	X	34.22°F

Notes:

(a) Table 4-4 is the same as Table 4-3 from Reference 13.

Table 4-5 contains the best estimate weight percent copper and nickel for the Byron Unit 2 base materials in the beltline region. Table 4-6 contains the calculation of the best estimate weight percent copper and nickel for the Byron Unit 2 surveillance weld material, while Table 4-7 presents the overall best estimate average for that heat of weld. Table 4-8 contains a summary of the weight percent of copper, the weight percent of nickel and the initial RT_{NDT} of the beltline materials and vessel flanges. The weight percent values of Cu and Ni given in Table 4-8 were used to generate the calculated chemistry factor (CF) values based on Tables 1 and 2 of Regulatory Guide 1.99, Revision 2, and presented in Table 4-11. Tables 4-9 and 4-10 provide the calculation of the CF values based on surveillance capsule data, Regulatory Guide 1.99, Revision 2, Position 2.1, which are also summarized in Table 4-11.

TABLE 4-5

Best Estimate Cu and Ni Weight Percent for the Byron Unit 2 Forging Materials

	Intermediate Shell Forging 49D329-1/49C297-1		**Lower Shell Forging 49D330-1/49C298-1**	
Reference	**Cu %**	**Ni %**	**Cu %**	**Ni %**
Best Estimate Average Calculated Per Reference 13	0.01	0.70	0.06	0.74

Note: The best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-6

Average Cu and Ni Weight Percent for the Byron Unit 2 Surveillance Weld Material Only (Heat # 442002)

Reference	**Weight % Copper**	**Weight % Nickel**
Surveillance Weld Average Calculated Per Reference 13	0.02	0.71

Note: The best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-7
Best Estimate Cu and Ni Weight Percent Values for the Byron Units 1 & 2
Weld Material (Using Byron 1 & 2 Chemistry Test Results)

Chemistry Type	Reference	Weight % Copper	Weight % Nickel
BEST ESTIMATE AVERAGE	13 & 14	0.04[(c)]	0.63[(c)]

NOTES:
(a) The weld material in the Byron Unit 1 surveillance program was made of the same wire and flux as the reactor vessel inter. to lower shell girth seam weld. (Weld seam WF-336, Wire Heat # 442002, Flux Type Linde 80, Flux Lot # 8873).
(b) The Byron Unit 2 surveillance weld is identical to that used in the reactor vessel core region girth seam (WF-447). The weld wire is type Linde MnMoNi (Low Cu-P), heat number 442002, with a Linde 80 type flux, lot number 8064.
(c) The best estimate chemistry values were obtained using the "average of averages" approach. In addition the best estimate average was rounded per ASTM E29, using the "Rounding Method".

TABLE 4-8[b]

Reactor Vessel Beltline Material Unirradiated Toughness Properties

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange 5P7382 / 3P6407	---	0.71	0
Vessel Flange 124L556VA1	---	0.70	30
Nozzle Shell Forging 4P-6107[b]	0.05	0.74	10
Intermediate Shell Forging 49D329-1/49C297-1	0.01	0.70	-20
Lower Shell Forging 49D330-1/49C298-1	0.06	0.74	-20
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat # 442002)	0.04	0.63	10
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-562 (Heat # 442011)	0.03	0.67	40
Byron Unit 1 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.69	---
Byron Unit 2 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.71	---
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03	0.67, 0.71	---

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) Table duplicated from Table 4-7 of Reference 13.

TABLE 4-9[a]

Calculation of Chemistry Factors using Byron Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule f	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Forging 49D330-1/49C298-1 (Tangential)	U	0.405	0.749	0.0[e]	0	0.561
	W	1.27	1.067	3.65	3.89	1.138
	X	2.30	1.225	15.75	19.29	1.500
Lower Shell Forging 49D330-1/49C298-1 (Axial)	U	0.405	0.749	19.76	14.80	0.561
	W	1.27	1.067	31.88	34.02	1.138
	X	2.30	1.225	38.91	47.66	1.500
				SUM:	119.66	6.398
	$CF_{Forging} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (119.66) \div (6.398) =$ **18.7°F**					
Byron Unit 1 Surv. Weld Material (Heat # 442002)	U	0.404	0.749	11.22 (5.61)[d]	8.40	0.561
	X	1.57	1.125	80.22 (40.11)[d]	90.25	1.266
	W	2.43	1.239	102.68 (51.34)[d]	127.22	1.535
Byron Unit 2 Surv. Weld Material (Heat # 442002)	U	0.405	0.749	16.88 (8.44)[d]	12.64	0.561
	W	1.27	1.067	57.76 (28.88)[d]	61.63	1.138
	X	2.30	1.225	108.02 (54.01)[d]	132.32	1.500
				SUM:	432.46	6.561
	$CF_{Surv.\ Weld,\ 442002} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (432.46) \div (6.561) =$ **65.9°F**					

Notes:

(a) The calculation for the Byron Unit 1 and 2 chemistry factor was taken from Reference 13 & 14

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (See Ref. 13 & 14).

(d) The Byron 1 & 2 surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 2.00. No temperature adjustment are required.

TABLE 4-10[a]

Calculation of Chemistry Factors using Braidwood Units 1 & 2 Surveillance Capsule Data

Material	Capsule	Capsule f	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Braidwood Unit 1	U	0.387	0.737	17.06[d]	12.57	0.543
Surveillance	X	1.24	1.060	30.15[d]	31.96	1.124
Weld Heat 442011, WF-501	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2	U	0.400	0.746	0.0[d, e]	0	0.557
Surveillance	X	1.23	1.058	26.3[d]	27.83	1.119
Weld Heat 442011, WF-501	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6.273) = 25.7°F$					

Notes:

(a) The calculation for the Byron Unit 1 and 2 chemistry factor was taken from Reference 15.

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (See Ref. 15)

(d) The Braidwood 1 & 2 surveillance weld metal ΔRT_{NDT} values do not require a ratio factor or temperature adjustment.

TABLE 4-11
Summary of the Byron Unit 2 Reactor Vessel Beltline Material Chemistry Factors
Based on Regulatory Guide 1.99, Revision 2, Position 1.1 and Position 2.1

Material	Chemistry Factor	
	Position 1.1	Position 2.1
Inter. Shell Forging 49D329-1/49C297-1	20.0°F	---
Lower Shell Forging 49D330-1/49C298-1	37.0°F	18.7°F
Nozzle Shell Forging 4P-6107	31.0°F	---
Intermediate Shell to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	54.0°F	65.9°F
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	41.0°F	25.7°F
Byron Unit 1 Surveillance Program Weld Metal	27.0°F	---
Braidwood Unit 1 & 2 Surveillance Weld Metal	41.0°F	

Contained in Tables 4-12 and 4-13 is the summary of the fluence factors (FF) used in the calculation of adjusted reference temperatures for the Byron Unit 2 reactor vessel beltline materials for 22 and 32 EFPY.

TABLE 4-12
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 22 EPFY Heatup/Cooldown Curves

Azimuth	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate. Shell Forging 49D329-1/49C297-1	8.47×10^{18}	0.953	3.05×10^{18}	0.675
Lower Shell Forging 49D330-1/49C298-1	8.47×10^{18}	0.953	3.05×10^{18}	0.675
Nozzle Shell Forging 4P-6107	2.15×10^{18}	0.587	7.75×10^{17}	0.368
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	8.35×10^{18}	0.949	3.01×10^{18}	0.671
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	2.15×10^{18}	0.587	7.75×10^{17}	0.368

TABLE 4-13
Calculation of the 1/4T and 3/4 T Fluence Factor Values used for the Generation of the 32 EPFY Heatup/Cooldown Curves

Azimuth	1/4 T F (n/cm^2, E > 1.0 MeV)	1/4T FF	3/4T F (n/cm^2, E >1.0 MeV)	3/4 T FF
Intermediate. Shell Forging 49D329-1/49C297-1	1.24×10^{19}	1.06	4.46×10^{18}	0.775
Lower Shell Forging 49D330-1/49C298-1	1.24×10^{19}	1.06	4.46×10^{18}	0.775
Nozzle Shell Forging 4P-6107	3.13×10^{18}	0.681	1.13×10^{18}	0.442
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	1.22×10^{19}	1.05	4.40×10^{18}	0.772
Nozzle Shell to Inter. Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	3.13×10^{18}	0.681	1.13×10^{18}	0.442

Contained in Tables 4-14 through 4-17 are the calculations of the ART values used for the generation of the 22 and 32 EFPY heatup and cooldown curves.

TABLE 4-14

Calculation of the ART Values for the 1/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 22[a] EFPY ($x 10^{19}$)	¼-t f($x 10^{19}$)	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	[49D329/ 49C297]-1-1	0.01	0.70	20.0	1.41	0.847	0.953	-20	19.1	0	9.5	19.1	18
Lower Shell Forging	[49D330/ 49C298]-1-1	0.06	0.74	37.0	1.41	0.847	0.953	-20	35.3	0	17.0	34.0	49
Lower shell Forging → using S/C Data				18.7	1.41	0.847	0.953	-20	17.8	0	8.5	17.0	15
Inter. to Lower Shell Circ. Weld Metal	WF-447	0.04	0.63	54.0	1.39	0.835	0.949	10	51.2	0	25.6	51.2	112
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.39	0.835	0.949	10	62.5	0	14.0	28.0	**101**
Nozzle Shell Forging	4P-6107	0.05	0.74	31.0	0.358	0.215	0.587	10	18.2	0	9.1	18.2	46
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.358	0.215	0.587	40	24.1	0	12.0	24.1	88
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.358	0.215	0.587	40	15.1	0	7.5	15.1	70

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

TABLE 4-15

Calculation of the ART Values for the 3/4T Location @ 22 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[(d)]	f @ 22[(a)] EFPY (x 10^{19})	¾-t f(x 10^{19})	¾-t FF	I	ΔRT_{NDT}[(c)]	σ_I	σ_Δ	M	ART[(b)]
Intermediate Shell Forging	[49D329/ 49C297]-1-1	0.01	0.70	20.0	1.41	0.305	0.675	-20	13.5	0	6.7	13.5	7
Lower Shell Forging	[49D330/ 49C298]-1-1	0.06	0.74	37.0	1.41	0.305	0.675	-20	25.0	0	12.5	25.0	30
Lower shell Forging → using S/C Data				18.7	1.41	0.305	0.675	-20	12.6	0	6.3	12.6	5
Inter. to Lower Shell Circ. Weld Metal	WF-447	0.04	0.63	54.0	1.39	0.301	0.671	10	36.2	0	18.1	36.2	**82**
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	1.39	0.301	0.671	10	44.2	0	14.0	28.0	**82**
Nozzle Shell Forging	4P-6107	0.05	0.74	31.0	0.358	0.0775	0.368	10	11.4	0	5.7	11.4	**33**
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.358	0.0775	0.368	40	15.1	0	7.5	15.1	70
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.358	0.0775	0.368	40	9.5	0	4.7	9.5	59

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

TABLE 4-16

Calculation of the ART Values for the 1/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 32[a] EFPY ($x 10^{19}$)	¼-t f($x 10^{19}$)	¼-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	[49D329/ 49C297]-1-1	0.01	0.70	20.0	2.06	1.24	1.060	-20	21.2	0	10.6	21.2	22
Lower Shell Forging	[49D330/ 49C298]-1-1	0.06	0.74	37.0	2.06	1.24	1.060	-20	39.2	0	17.0	34.0	53
Lower shell Forging → using S/C Data				18.7	2.06	1.24	1.060	-20	19.8	0	8.5	17.0	17
Inter. to Lower Shell Circ. Weld Metal	WF-447	0.04	0.63	54.0	2.03	1.22	1.05	10	56.7	0	28.0	56.0	123
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	2.03	1.22	1.05	10	69.2	0	14.0	28.0	107
Nozzle Shell Forging	4P-6107	0.05	0.74	31.0	0.522	0.313	0.681	10	21.1	0	10.6	21.1	52
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.522	0.313	0.681	40	27.9	0	14.0	27.9	96
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.522	0.313	0.681	40	17.5	0	8.8	17.5	75

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

TABLE 4-17

Calculation of the ART Values for the 3/4T Location @ 32 EFPY

Reactor Vessel Beltline Region Location	Material Identification	Cu%	Ni%	CF[d]	f @ 32[a] EFPY (x 10^{19})	¾-t f(x 10^{19})	¾-t FF	I	ΔRT_{NDT}[c]	σ_I	σ_Δ	M	ART[b]
Intermediate Shell Forging	[49D329/ 49C297]-1-1	0.01	0.70	20.0	2.06	.446	.775	-20	15.5	0	7.8	15.5	11
Lower Shell Forging	[49D330/ 49C298]-1-1	0.06	0.74	37.0	2.06	.446	.775	-20	28.7	0	14.3	28.7	37
Lower shell Forging → using S/C Data				18.7	2.06	.446	.775	-20	14.5	0	7.2	14.5	9
Inter. to Lower Shell Circ. Weld Metal	WF-447	0.04	0.63	54.0	2.03	.440	.772	10	41.7	0	20.8	41.7	93
Inter. to Lower Shell Circ. Weld Metal → using S/C Data				65.9	2.03	.440	.772	10	50.9	0	14.0	28.0	89
Nozzle Shell Forging	4P-6107	0.05	0.74	31.0	0.522	0.113	0.442	10	13.7	0	6.9	13.7	37
Nozzle Shell to Inter. Shell Circ. Weld Metal	WF-562	0.03	0.67	41.0	0.522	0.113	0.442	40	18.1	0	9.1	18.1	76
Nozzle Shell to Inter. Shell Circ. Weld Metal → using S/C Data				25.7	0.522	0.113	0.442	40	11.4	0	5.7	11.4	63

NOTES:

(a) Fluence, f, is based upon f_{surf} (10^{19} n/cm^2, E>1.0 MeV).

(b) ART = I + ΔRT_{NDT} + M (This value was rounded per ASTM E29, using the "Rounding Method".)

(c) ΔRT_{NDT} = CF * FF

(d) The CF for the Inter. to Lower Shell Circ. Weld is integrated between the Byron 1 Weld (WF-336, heat # 442002) and the Byron 2 Weld (WF-447, Heat # 442002).
The CF for the Nozzle Shell to Inter. Shell Circ. Weld is integrated between Byron 1 Weld (WF-501, heat # 442011) and the Braidwood 1 & 2 Welds (WF-562, heat # 442011).

The girth weld WF-447 and the nozzle shell forging 4P-6107 are the limiting beltline materials for all heatup and cooldown curves to be generated. The ART value associated with these materials will be used in all three sets of curves. The girth weld ART will be used when generating curves for Code Case N-588 (ie. Circ. Flaw), otherwise, the nozzle shell forging ART will be used. The ART associated with the limiting axial material is considered to determine if this case would be more conservative or overlap the circ. flaw curves. Contained in Tables 4-18 and 4-19 is a summary of the limiting ARTs to be used in the generation of the Byron Unit 2 reactor vessel heatup and cooldown curves.

TABLE 4-18

Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4T Location for 22 EFPY

Material	**22 EFPY**	
	1/4T ART	**3/4T ART**
Intermediate Shell Forging [49D329/49C297]-1-1	18	7
Lower Shell Forging [49D330/49C298]-1-1	49	30
- Using Surveillance Data[a]	15	5
Circumferential Weld WF-447	112	**82**
- Using Surveillance Data	**101**	**82**
Circumferential Weld WF-562	88	70
- Using Surveillance Data from Braidwood 1 and 2	70	59
Nozzle Shell Forging 4P-6107	**46[a]**	**33[a]**

NOTES:

(a) These ART values were used to calculate the Heatup and cooldown curves in Figure 5-1 and 5-2. They were generated using the '96 App. G Methodology, and were more conservative than the curves generated using the limiting circumferential flaw ART values and Code Case N-588 Methodology.

TABLE 4-19

Summary of Adjusted Reference Temperature (ART) at 1/4T and 3/4T Location for 32 EFPY

Material	32 EFPY	
	1/4T ART	**3/4T ART**
Intermediate Shell Forging [49D329/49C297]-1-1	22	11
Lower Shell Forging [49D330/49C298]-1-1	53	37
- Using Surveillance Data[a]	17	9
Circumferential Weld WF-447	123	93
- Using Surveillance Data	**107**	**89**
Circumferential Weld WF-562	96	76
- Using Surveillance Data from Braidwood 1 and 2	75	63
Nozzle Shell Forging 4P-6107	**52[a]**	**37[a]**

NOTES:

(a) These ART values were used to calculate the Heatup and cooldown curves in Figure 5-3 and 5-4. They were generated using the '96 App. G Methodology, and were more conservative than the curves generated using the limiting circumferential flaw ART values and Code Case N-588 Methodology.

5 HEATUP AND COOLDOWN PRESSURE-TEMPERATURE LIMIT CURVES

Pressure-temperature limit curves for normal heatup and cooldown of the primary reactor coolant system have been calculated for the pressure and temperature in the reactor vessel beltline region using the methods discussed in Section 3 and 4 of this report. This approved methodology is also presented in WCAP-14040-NP-A[8], dated January 1996.

Figures 5-1 through 5-4 present the 22 and 32 EFPY heatup and cooldown curves (without margins for possible instrumentation errors) for a heatup rate of 100°F/hr and cooldown rates of 0, 25, 50 and 100°F/hr using the 1996 Appendix G methodology[6] and Code Case N-588[5], respectively. The heatup and cooldown curves that are presented herein are actually are curves generated using the 1996 App. G methodology with the *lower axial flaw ART value* with exception to the last four temperatures of the 32 EFPY 100°F/hr heatup curve. The reason is, these curves are more conservative (with exception noted) than the curves generated using Code Case N-588 methodology with the *higher circ. flaw ART value*. This is true throughout the entire temperature range, including criticality.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown in Figures 5-1 through 5-4. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figures 5-1and 5-3 (for the specific heatup rate being utilized). The straight-line portion of the criticality limit is at the minimum permissible temperature for the 2485 psig inservice hydrostatic test as required by Appendix G to 10 CFR Part 50. The governing equation for the hydrostatic test is defined in Code Case N-640[2] (approved in February 1999) as follows:

$$1.5K_{Im} < K_{Ic} \qquad (11)$$

where,

K_{Im} is the stress intensity factor covered by membrane (pressure) stress,

$K_{Ic} = 33.2 + 20.734\, e^{[0.02\,(T - RT_{NDT})]}$,

T is the minimum permissible metal temperature, and

RT_{NDT} is the metal reference nil-ductility temperature

The criticality limit curve specifies pressure-temperature limits for core operation to provide additional margin during actual power production as specified in Reference 3. The pressure-temperature 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 temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure-temperature curve for heatup and cooldown calculated as described in Section 3 of this report. The minimum temperatures for the inservice hydrostatic leak test for the Byron Unit 2 reactor vessel at 22 and 32 EFPY are 106°F and

112°F at 2485 psig 1996 App. G Methodology. The vertical line drawn from these points on the pressure-temperature curve, intersecting a curve 40°F higher than the pressure-temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 5-1 through 5-4 define all of the above limits for ensuring prevention of nonductile failure for the Byron Unit 2 reactor vessel. The data points for the heatup and cooldown pressure-temperature limit curves shown in Figures 5-1 through 5-4 are presented in Tables 5-1 through 5-4, respectively.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING

LIMITING ART VALUES AT 22 EFPY: 1/4T, 101°F (N-588) & 46°F ('96 App. G)

3/4T, 82°F (N-588) & 33°F ('96 App. G)

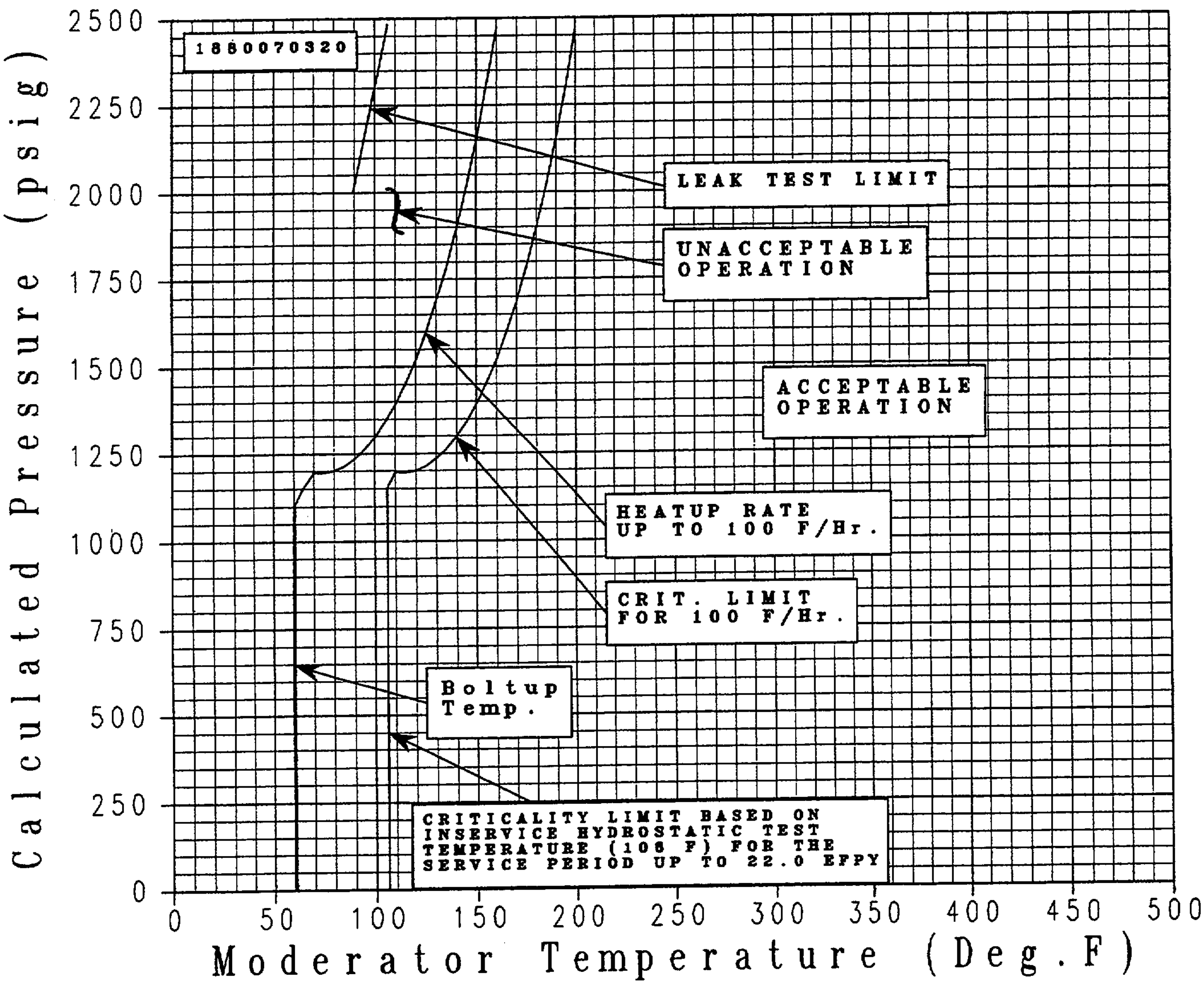


FIGURE 5-1 Byron Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 22 EFPY Using Code Case N-640 & 1996 Appendix G (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 22 EFPY: 1/4T, 101°F (N-588) & 46°F ('96 App. G)
3/4T, 82°F (N-588) & 33°F ('96 App. G)

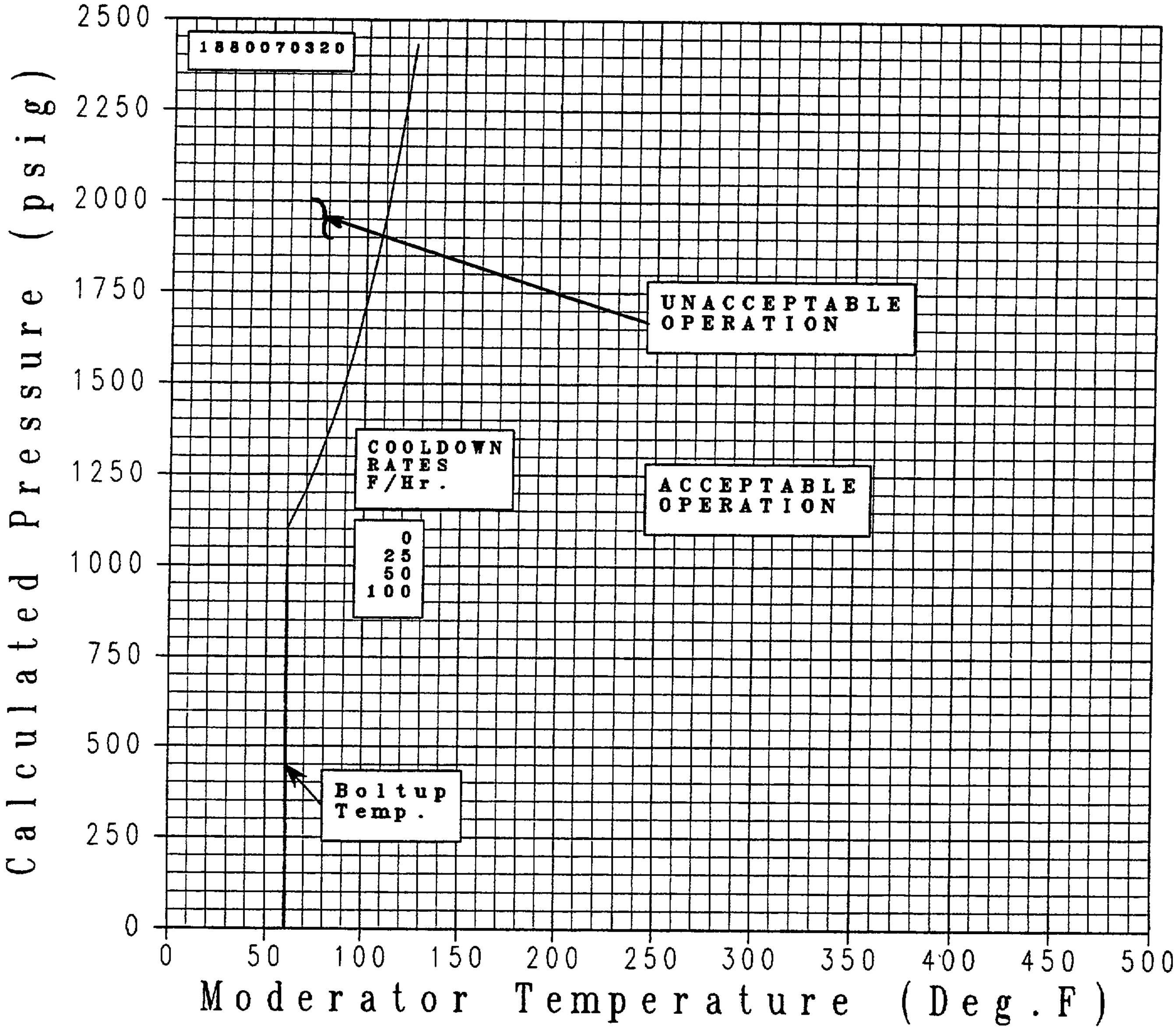


FIGURE 5-2 Byron Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 22 EFPY Using Code Case N-640 & 1996 Appendix G (Without Margins of for Instrumentation Errors)

TABLE 5-1

Byron Unit 2 Heatup Data at 22 EFPY Using Code Case N-640 & 1996 Appendix G Methodology*

(Without Margins for Instrumentation Errors)

Heatup Curves					
100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	0	106	0	89	2000
60	1102	106	1154	106	2485
65	1154	110	1199		
70	1197	115	1197		
75	1197	120	1202		
80	1202	125	1215		
85	1215	130	1235		
90	1235	135	1263		
95	1263	140	1299		
100	1299	145	1342		
105	1342	150	1393		
110	1393	155	1452		
115	1452	160	1520		
120	1520	165	1598		
125	1598	170	1686		
130	1686	175	1784		
135	1784	180	1895		
140	1895	185	2018		
145	2018	190	2155		
150	2155	195	2304		
155	2304	200	2428		
160	2428				

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall with exception to the last two temperatures.

TABLE 5-2
Byron Unit 2 Cooldown Data at 22 EFPY Using Code Case N-640 & 1996 App. G Methodology*
(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	1102						
65	1154						
70	1212						
75	1276						
80	1347						
85	1425						
90	1512						
95	1607						
100	1713						
105	1829						
110	1958						
115	2101						
120	2258						
125	2433						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 107°F (N-588) & 52°F ('96 App. G)
3/4T, 89°F (N-588) & 37°F ('96 App. G)

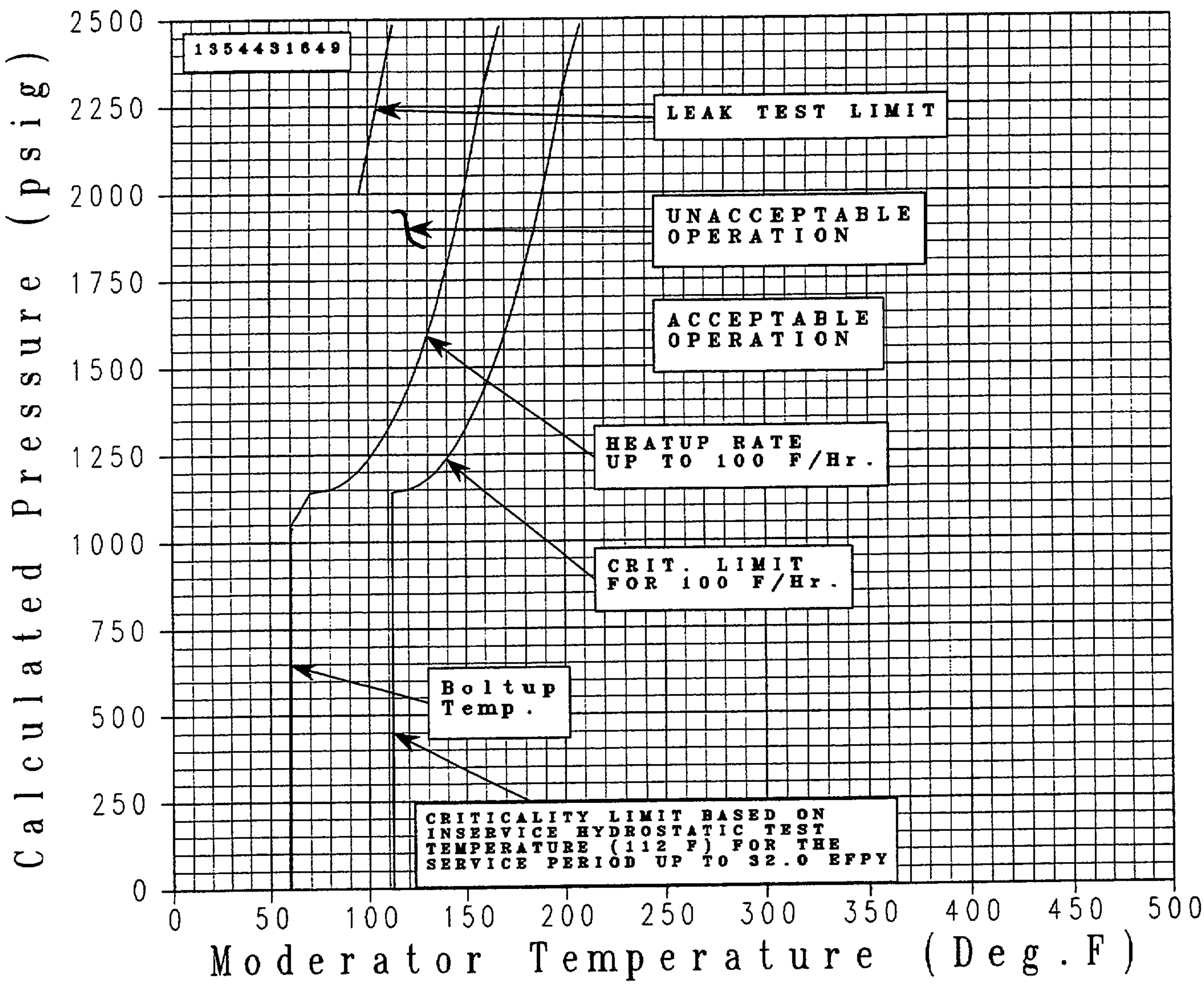


FIGURE 5-3 Byron Unit 2 Reactor Coolant System Heatup Limitations (Heatup Rate of 100°F/hr) Applicable to 32 EFPY Using Code Case N-640 & 1996 Appendix G (Without Margins of for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: CIRCUMFERENTIAL WELD WF-447 & NOZZLE SHELL FORGING
LIMITING ART VALUES AT 32 EFPY: 1/4T, 107°F (N-588) & 52°F ('96 App. G)
3/4T, 89°F (N-588) & 37°F ('96 App. G)

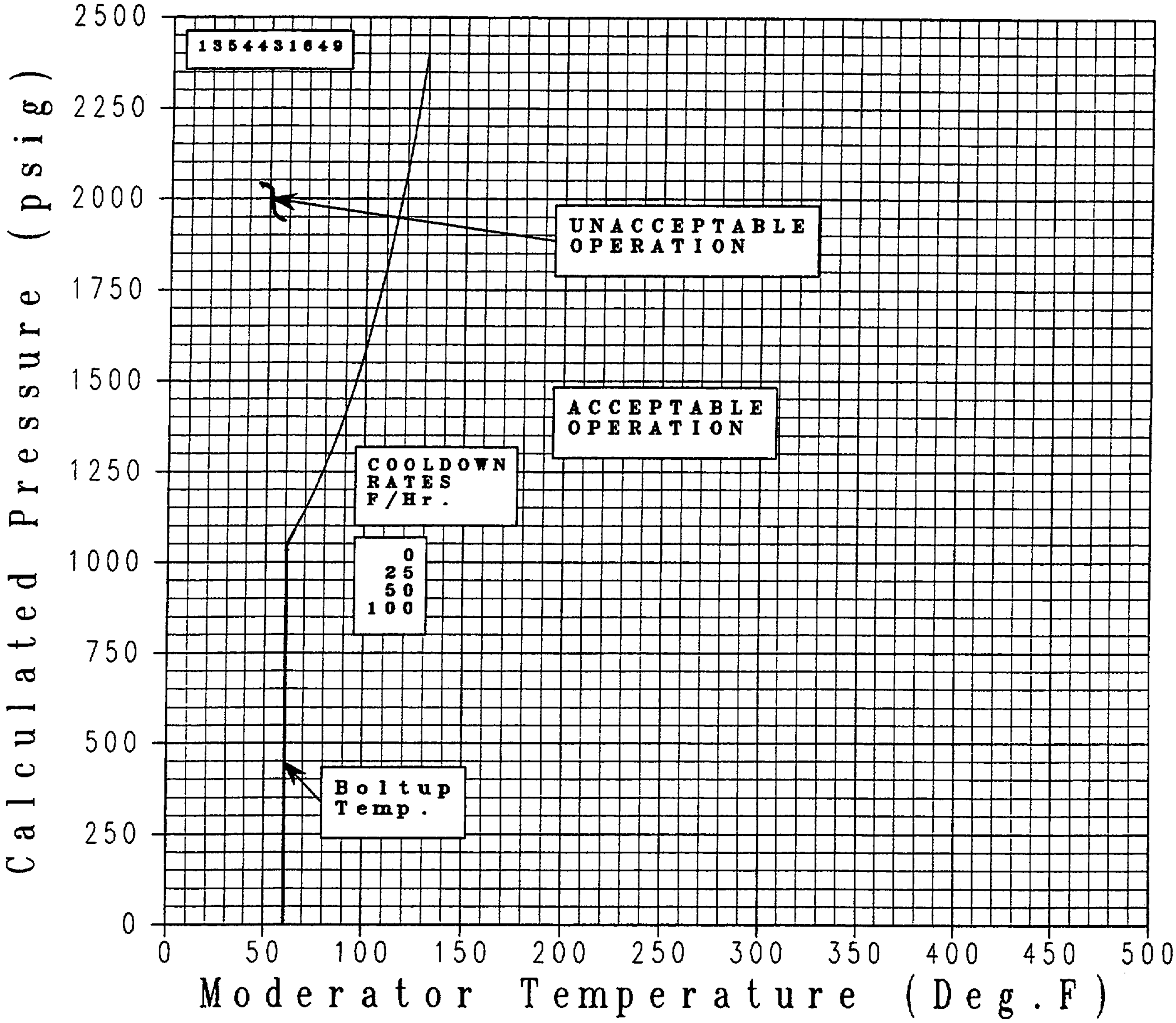


FIGURE 5-4 Byron Unit 2 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100°F/hr) Applicable to 32 EFPY Using Code Case N-640 & 1996 Appendix G (Without Margins of for Instrumentation Errors)

TABLE 5-3
Byron Unit 2 Heatup Data at 32 EFPY Using Code Case N-640 & 1996 App. G Methodology*
(Without Margins for Instrumentation Errors)

Heatup Curves					
100 Heatup		Critical. Limit		Leak Test Limit	
T	P	T	P	T	P
60	1045	112	0	95	2000
65	1092	112	1092	112	2485
70	1143	112	1143		
75	1148	115	1148		
80	1152	120	1152		
85	1162	125	1162		
90	1180	130	1180		
95	1205	135	1205		
100	1237	140	1237		
105	1276	145	1276		
110	1323	150	1323		
115	1377	155	1377		
120	1440	160	1440		
125	1511	165	1511		
130	1592	170	1592		
135	1682	175	1682		
140	1784	180	1784		
145	1897	185	1897		
150	2023	190	2023		
155	2120	195	2120		
160	2227	200	2227		
165	2347	205	2347		
170	2480	210	2480		

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall with exception to the last four temperatures.

TABLE 5-4
Byron Unit 2 Cooldown Data at 32 EFPY Using Code Case N-640 & 1996 App. G Methodology*
(Without Margins for Instrumentation Errors)

Cooldown Curves							
Steady State		25F		50F		100F	
T	P	T	P	T	P	T	P
60	0	60	0	60	0	60	0
60	1045	60	1036	60	1033		
65	1092	65	1088	65	1092		
70	1143						
75	1200						
80	1263						
85	1332						
90	1409						
95	1494						
100	1587						
105	1691						
110	1805						
115	1932						
120	2071						
125	2226						
130	2396						

* Note: The computer run for the '96 App. G using the highest Axial Flaw ART value generated the most conservative curve overall.

6 REFERENCES

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

2. ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limit Curves for Section XI, Division 1", February 26, 1999.

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

4. WCAP-15315, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation For Operating PWR and BWR Plants", W. Bamford, et.al., October 1999.

5. ASME Code Case N-588, "Attenuation to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels", Section XI, Division 1, Approved December 12, 1997.

6. 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.

7. WCAP-15183, "Commonwealth Edison Company Byron Unit 1 Surveillance Program Credibility Evaluation", T.J. Laubham, June 1999.

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

9. 1989 Section III, Division 1 of the ASME Boiler and Pressure Vessel Code, Paragraph NB-2331, "Material for Vessels".

10. SAE-REA-00-546, "Reactor Vessel Neutron Exposure Projections for the Byron/Braidwood Uprating", J.D. Perock, January 12, 2000.

11. WCAP-15180, "Commonwealth Edison Company Byron Unit 2 Surveillance Program Credibility Evaluation", T.J. Laubham, June 1999.

12. WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance Weld Metal Integration For Byron and Braidwood", T. J. Laubham, et al., November 1997. Ref. Errata letter CAE-97-233, CCE-97-316, "Transmittal of Updated Tables to WCAP-14824 Rev. 2"

13. WCAP-15178, "Byron Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, June 1999.

14 WCAP-15124, "Byron Unit 1 Heatup and Cooldown for Normal Operation", T. J. Laubham, November 1998.

15 WCAP-15373, "Braidwood Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, March 2000.

16 CVGRAPH, Hyperbolic Tangent Curve-Fitting Program, Version 4.1, developed by ATI Consulting, March 1996.

17 WCAP-15368, "Commonwealth Edison Company Braidwood Unit 2 Surveillance Program Credibility Evaluation", T.J. Laubham, March 2000.

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15390

Evaluation of Pressurized Thermal Shock for Byron Unit 1

T.J. Laubham

September 2000

Prepared by the Westinghouse Electric Company LLC
for the Commonwealth Edison Company

Approved: C H Boyd
C. H. Boyd, Manager
Engineering & Materials Technology

Approved: ____________________
D. M. Trombola, Manager
Mechanical Systems Integration

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

PREFACE

This report has been technically reviewed and verified by:

Reviewer: Ed Terek *Ed Terek*

EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the Byron Unit 1 reactor vessel beltline based upon the 5% uprated fluence values. The conclusion of this report is that all the beltline materials in the Byron Unit 1 reactor vessel have RT_{PTS} values below the screening criteria of 270°F for plates, forgings or longitudinal welds and 300°F for circumferential welds at EOL (32 EFPY) and life extension (48 EFPY). Specifically, the intermediate shell forging was the most limiting material with 32 and 48 EFPY PTS values of 110°F and 113°F respectively.

1 INTRODUCTION

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce a flaw or cause the propagation of a flaw postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The purpose of this report is to determine the RT_{PTS} values for the Byron Unit 1 reactor vessel using the fluence results of the 5% uprating and the previous PTS Report in WCAP-15125[8]. Section 2.0 discusses the PTS Rule and its requirements. Section 3.0 provides the methodology for calculating RT_{PTS}. Section 4.0 provides the reactor vessel beltline region material properties for the Byron Unit 1 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from SAE-REA-00-546[9]. The results of the RT_{PTS} calculations are presented in Section 6.0. The conclusion and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

2 PRESSURIZED THERMAL SHOCK RULE

The Nuclear Regulatory Commission (NRC) amended its regulations for light-water-cooled nuclear power plants to clarify several items related to the fracture toughness requirements for reactor pressure vessels, including pressurized thermal shock requirements. The latest revision of the PTS Rule[1], 10 CFR Part 50.61, was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996.

This amendment to the PTS Rule makes three changes:

1. The rule incorporates in total, and therefore makes binding by rule, the method for determining the reference temperature, RT_{NDT}, including treatment of the unirradiated RT_{NDT} value, the margin term, and the explicit definition of "credible" surveillance data, which is also described in Regulatory Guide 1.99, Revision 2[2].

2. The rule is restructured to improve clarity, with the requirements section giving only the requirements for the value for the reference temperature for end of license (EOL) fluence, RT_{PTS}.

3. Thermal annealing is identified as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}.

The PTS Rule requirements consist of the following:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule, and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewal term, if applicable for the plant.

- The RT_{PTS} screening criterion values for the beltline region are:

  270°F for plates, forgings and axial weld materials, and

  300°F for circumferential weld materials.

3 METHOD FOR CALCULATION OF RT_{PTS}

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld and plate or forging in the reactor vessel beltline.

$$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT} \quad (1)$$

Where,

$RT_{NDT(U)}$ = Reference Temperature for a reactor vessel material in the pre-service or unirradiated condition

M = Margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures. M is evaluated from Equation 2

$$M = \sqrt{\sigma_U^2 + \sigma_\Delta^2} \quad (2)$$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value.

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value.

σ_Δ is the standard deviation for RT_{NDT}.

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used.

σ_Δ = 8.5°F when surveillance capsule data is used.

For welds:

σ_Δ = 28°F when surveillance capsule data is not used.

σ_Δ = 14°F when surveillance capsule data is used.

σ_Δ not to exceed one half of ΔRT_{NDT}

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in ΔRT_{NDT}, due to irradition, and must be calculated using Equation 3.

$$\Delta RT_{NDT} = (CF) * f^{(0.28-0.10\log f)} \quad (3)$$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

The EOL Fluence (f) is the higher of the best estimate or calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The EOL fluence is used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining RT_{PTS}..

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS} \qquad (4)$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible.

A material-specific value of CF for surveillance materials is determined from Equation 5.

$$CF = \frac{\sum [A_i * f_i^{(0.28-0.10 \log f_i)}]}{\sum [f_i^{(0.56-0.20 \log f_i)}]} \qquad (5)$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

4 VERIFICATION OF PLANT SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties for the Byron Unit 1 vessel was performed. The beltline region of a reactor vessel, per the PTS Rule, is defined as "the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage". Figure 1 identifies and indicates the location of all beltline region materials for the Byron Unit 1 reactor vessel.

The best estimate copper and nickel contents of the beltline materials were originally obtained from WCAP-14824, Revision 2[4] and from the testing of Charpy specimens in Capsule W[5]. The best estimate copper and nickel content is documented in Table 1 herein. The average values were calculated using all of the available material chemistry information. Initial RT_{NDT} values for Byron Unit 1 reactor vessel beltline material properties are also shown in Table 1.

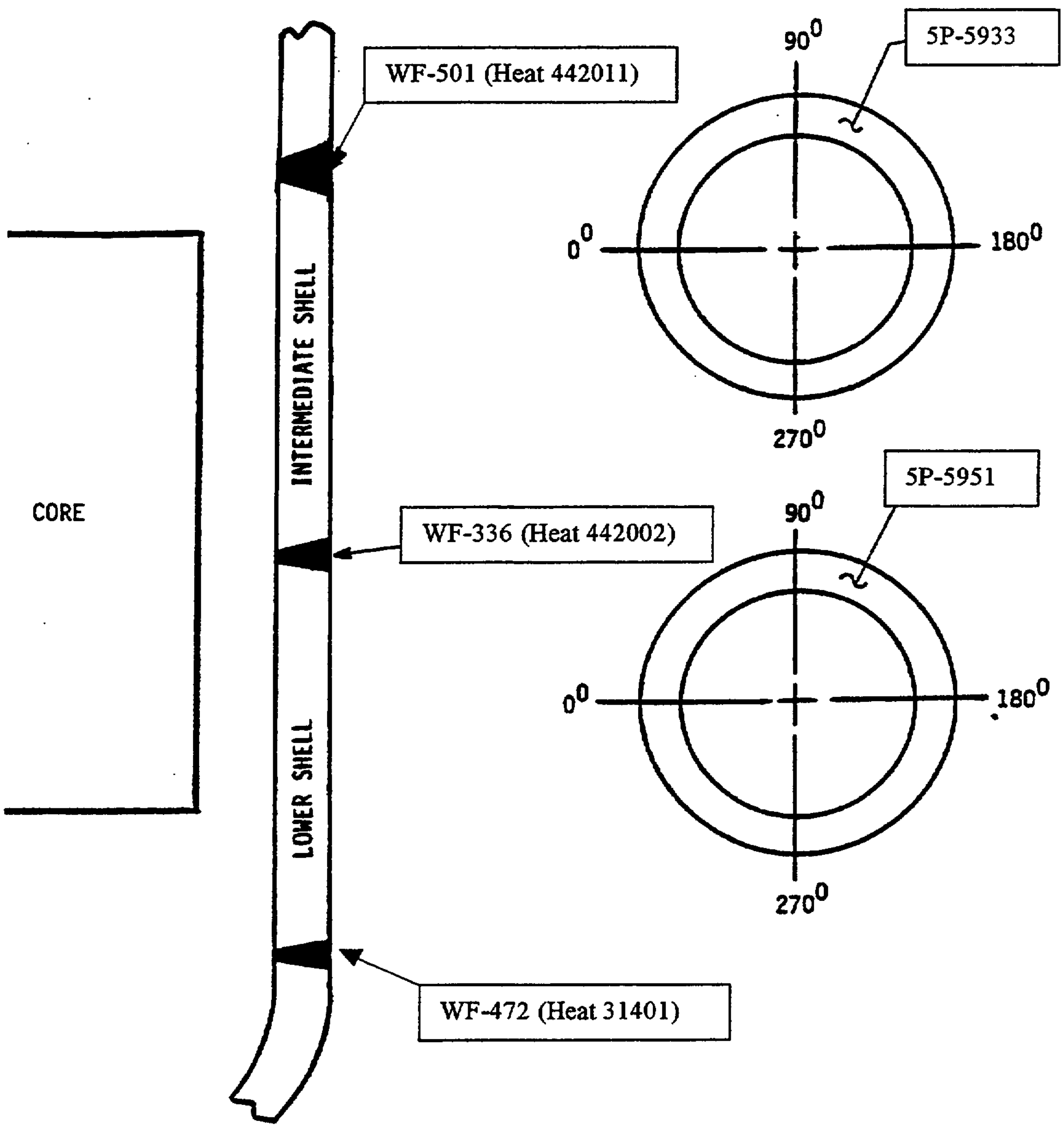


Figure 1: Identification and Location of Beltline Region Materials for the Byron Unit 1 Reactor Vessel

Table 1[b]

Byron Unit 1 Reactor Vessel Beltline Unirradiated Material Properties

Material Description	Cu (%)	Ni(%)	Initial RT_{NDT}[a]
Closure Head Flange 124K358VA1	---	0.74	60
Vessel Flange 123J219VA1	---	0.73	10
Nozzle Shell Forging 123J218[b]	0.05	0.72	30
Intermediate Shell Forging 5P-5933	0.04	0.74	40
Lower Shell Forging 5P-5951	0.04	0.64	10
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat # 442002)	0.04	0.63	-30
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-501 (Heat # 442011)[b]	0.03	0.67	10
Byron Unit 1 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.69	---
Byron Unit 2 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.71	---
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03	0.67, 0.71	---

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) Table duplicated from Table 1 of Reference 8.

5 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the inner surface of the Byron Unit 1 reactor vessel for 32 and 48 EFPY are shown in Table 2. These values were projected using the fluence results of the 5% uprating analysis[9].

TABLE 2

Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Interface for Byron Unit 1 at 32 (EOL) and 48 (Life Extension) EFPY

Material	Location	32 EFPY Fluence	48 EFPY Fluence
Nozzle Shell Forging 123J218	45°	6.04×10^{18} n/cm^2	0.91×10^{19} n/cm^2
Intermediate Shell Forging 5P-5933	45°	2.02×10^{19} n/cm^2	3.03×10^{19} n/cm^2
Lower Shell Forging 5P-5951	45°	2.02×10^{19} n/cm^2	3.03×10^{19} n/cm^2
Intermediate to Lower Shell Forging Circ. Weld Seam WF-336 (Heat # 442002)	45°	1.94×10^{19} n/cm^2	2.91×10^{19} n/cm^2
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-501 (Heat # 442011)	45°	6.04×10^{18} n/cm^2	0.91×10^{19} n/cm^2

6 DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIALS

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Byron Unit 1 reactor vessel for fluence values at the EOL (32 EFPY) and life extension (48 EFPY).

Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. For Byron Unit 1, the related surveillance program results have been included in this PTS evaluation. Per Reference 10, the surveillance weld data for Byron Unit 1 is credibile while the surveillance forging material is non-credible. In addition, Reference 10 also shows that the Table chemistry factor is conservative. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine the RT_{PTS} of the forging material with a full margin term.

As presented in Table 3, chemistry factor values for Byron Unit 1 based on average copper and nickel weight percent were calculated using Tables 1 and 2 from 10 CFR 50.61[1]. Additionally, chemistry factor values based on surveillance capsule data from Byron Units 1 and 2, and Braidwood Units 1 and 2 are calculated in Table 4. Tables 5 and 6 contain the RT_{PTS} calculations for all beltline region materials at EOL (32 EFPY) and life extension (48 EFPY).

TABLE 3

Interpolation of Chemistry Factors Using Tables 1 and 2 of 10 CFR Part 50.61

Material	Ni, wt %	Chemistry Factor, °F
Intermediate Shell Forging 5P-5933 Given Cu wt% = 0.04	0.74	26.0°F
Lower Shell Forging 5P-5951 Given Cu wt% = 0.04	0.64	26.0°F
Nozzle Shell Forging 123J218 Given Cu wt% = 0.05	0.72	31.0°F
Intermediate to Lower Shell Circ. Weld WF-336 Given Cu wt% = 0.04	0.63	54.0°F
Nozzle Shell to Intermediate Shell Circ. Weld WF-501 Given Cu wt% = 0.03	0.67	41.0°F
Byron Unit 1 and 2 Surveillance Program Weld Metal Given Cu wt% = 0.02	0.69, 0.71	27.0°F
Braidwood Unit 1 and 2 Surveillance Program Weld Metal Given Cu wt% = 0.03	0.67, 0.71	41.0°F

TABLE 4a

Calculation of Chemistry Factors using Byron Unit 1 Surveillance Capsule Data

Material	Capsule	Capsule f	FF(b)	ΔRT_{NDT}(c)	$FF*\Delta RT_{NDT}$	FF^2
Intermediate Shell Forging 5P-5933 (Tangential)	U	0.404	0.748	28.55	21.36	0.560
	X	1.57	1.124	9.82	11.04	1.263
	W	2.43	1.239	49.20	60.96	1.535
Intermediate Shell Forging 5P-5933 (Axial)	U	0.404	0.748	18.52	13.85	0.560
	X	1.57	1.124	53.03	59.61	1.263
	W	2.43	1.239	29.34	36.35	1.535
				SUM:	203.17	6.716
	$CF_{5P\text{-}5933} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (203.17) \div (6.716) =$ **30.3°F**					
Byron Unit 1 Surv. Weld Material (Heat # 442002)	U	0.404	0.749	11.22 (5.61)(d)	8.40	0.561
	X	1.57	1.125	80.22 (40.11)(d)	90.25	1.266
	W	2.43	1.239	102.68 (51.34)(d)	127.22	1.535
Byron Unit 2 Surv. Weld Material (Heat # 442002)	U	0.405	0.749	16.88 (8.44)(d)	12.64	0.561
	W	1.27	1.067	57.76 (28.88)(d)	61.63	1.138
	X	2.30	1.225	108.02 (54.01)(d)	132.32	1.500
				SUM:	432.46	6.561
	$CF_{Surv.\ Weld,\ 442002} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (432.46) \div (6.561) =$ **65.9°F**					

Notes:

(a) The calculation for the Byron Unit 1 and 2 chemistry factors were taken from Reference 11 & 12

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (See Ref. 11 & 12).

(d) The Byron 1 & 2 surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 2.00. No temperature adjustments are required.

TABLE 4b

Calculation of Chemistry Factors using Braidwood Units 1 & 2 Surveillance Capsule Data

Material	Capsule	Capsule f	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Braidwood Unit 1	U	0.387	0.737	17.06[d]	12.57	0.543
Surveillance	X	1.24	1.060	30.15[d]	31.96	1.124
Weld Heat 442011, WF-501	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2	U	0.400	0.746	0.0[d]	0	0.557
Surveillance	X	1.23	1.058	26.3[d]	27.83	1.119
Weld Heat 442011, WF-501	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	$CF = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (161.19) \div (6.273) =$ **25.7°F**					

Notes:

(a) The calculation for the Braidwood Units 1 and 2 chemistry factor was taken from Reference 13.

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (See Ref. 13)

(d) The Braidwood 1 & 2 surveillance weld metal ΔRT_{NDT} values do not require a ratio factor or temperature adjustment.

TABLE 5
RT_{PTS} Calculation for Byron Unit 1 Beltline Region Materials at EOL (32 EFPY)

Material	Fluence, x 10^{19} (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[c] (°F)	Margin (°F)	$RT_{NDT(U)}$[a] (°F)	RT_{PTS}[b] (°F)
Intermediate Shell Forging	2.02	1.19	26	30.9	30.9	40	102
Intermediate Shell Forging (Using S/C Data)[d]	2.02	1.19	30.3	36.1	34	40	110
Lower Shell Forging	2.02	1.19	26	30.9	30.9	10	72
Nozzle Shell Forging 4P-6107	0.604	0.86	31	26.7	26.7	30	83
Inter. to Lower Shell Circ. Weld Metal	1.94	1.18	54	63.7	56	-30	90
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	1.94	1.18	65.9	77.8	28	-30	76
Nozzle Shell to Inter. Shell Circ. Weld Metal	0.604	0.86	41	35.3	35.3	10	81
Nozzle Shell to Inter. Shell Circ. Weld Metal (Using S/C Data)	0.604	0.86	25.7	22.1	22.1	10	54

Notes:

(a) Initial RT_{NDT} values are measured values (See Table 1)

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)

(c) $\Delta RT_{PTS} = CF * FF$

(d) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine the RT_{PTS} with a full σ_Δ margin term.

TABLE 6

RT_{PTS} Calculation for Byron Unit 1 Beltline Region Materials at Life Extension (48 EFPY)

Material	Fluence, x 10^{19} (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS} [c] (°F)	Margin (°F)	$RT_{NDT(U)}$ [a] (°F)	RT_{PTS} [b] (°F)
Intermediate Shell Forging	3.03	1.29	26	33.5	33.5	40	107
Intermediate Shell Forging (Using S/C Data)	3.03	1.29	30.3	39.1	34	40	113
Lower Shell Forging	3.03	1.29	26	33.5	33.5	10	77
Nozzle Shell Forging 4P-6107	0.91	0.97	31	30.1	30.1	30	90
Inter. to Lower Shell Circ. Weld Metal	2.91	1.28	54	69.1	56	-30	95
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	2.91	1.28	65.9	84.4	28	-30	82
Nozzle Shell to Inter. Shell Circ. Weld Metal	0.91	0.97	41	39.8	39.8	10	90
Nozzle Shell to Inter. Shell Circ. Weld Metal (Using S/C Data)	0.91	0.97	25.7	24.9	24.9	10	60

Notes:

(a) Initial RT_{NDT} values are measured values (See Table 1)

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)

(c) $\Delta RT_{PTS} = CF * FF$

(d) Surveillance data is considered not credible. In addition, the Table chemistry factor is conservative and would normally be used for calculating RT_{PTS}. However, because the chemistry factor predicted by the Regulatory Guide 1.99 Position 2.1 for the forging surveillance data was greater than the Position 1.1 chemistry factor, then the Position 2.1 chemistry factor will be used to determine the RT_{PTS} with a full σ_{Δ} margin term.

7 CONCLUSIONS

As shown in Tables 5 and 6, all of the beltline region materials in the Byron Unit 1 reactor vessel have EOL (32 EFPY) RT_{PTS} and Life Extension (48 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates, forgings and longitudinal welds and 300°F for circumferential welds. Specifically, the intermediate shell forging was the most limiting material with 32 and 48 EFPY PTS values of 110°F and 113°F respectively.

8 REFERENCES

1 10 CFR Part 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events", Federal Register, Volume 60, No. 243, dated December 19, 1995.

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

3 WCAP-9517, "Commonwealth Edison Co. Byron Station Unit No. 1 Reactor Vessel Radiation Surveillance Program", J. A. Davidson, July 1979.

4 WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance weld Metal Integration for Byron & Braidwood", T. J.Laubham, et. al., November, 1997.

5 WCAP-15123, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program", T.J. Laubham, et. al., January 1999.

6 Letters CAE-97-231, CCE-97-314, "ComEd Response to NRC Question to WCAP-14940, WCAP-14970 and 14824 Rev. 2", From C.S. Hauser to Mr. Guy DeBoo (of ComEd), Dated January 6, 1998.

7. ComEd Letter to U.S. Regulatory Commission, "Response to Additional Information Regarding Reactor Pressure Vessel", From R.M. Krich, Dated September 3, 1998.

8. WCAP-15125, "Evaluation of Pressurized Thermal Shock for Byron Unit 1", T.J. Laubham, November 1998.

9. SAE-REA-00-546, "Reactor Vessel Neutron Exposure Projections for the Byron/Braidwood Uprating", J.D. Perock, January 12, 2000.

10. WCAP-15183, "Commonwealth Edison Company Byron Unit 1 Surveillance Program Credibility Evaluation", T.J. Laubham, June 1999.

11. WCAP-15124, "Byron Unit 1 Heatup and Cooldown for Normal Operation", T. J. Laubham, November 1998.

12. WCAP-15178, "Byron Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, June 1999.

13. WCAP-15373, "Braidwood Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, March 2000.

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-15389

Evaluation of Pressurized Thermal Shock for Byron Unit 2

T.J. Laubham

September 2000

Prepared by the Westinghouse Electric Company LLC
for the Commonwealth Edison Company

Approved: C H Boyd
C. H. Boyd, Manager
Engineering & Materials Technology

Approved: [signature]
D. M. Trombola, Manager
Mechanical Systems Integration

Westinghouse Electric Company LLC
Energy Systems
P.O. Box 355
Pittsburgh, PA 15230-0355

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

PREFACE

This report has been technically reviewed and verified by:

Reviewer: Ed Terek

EXECUTIVE SUMMARY

The purpose of this report is to determine the RT_{PTS} values for the Byron Unit 2 reactor vessel beltline based upon the 5% uprated fluence values. The conclusion of this report is that all the beltline materials in the Byron Unit 2 reactor vessel have RT_{PTS} values below the screening criteria of 270°F for plates, forgings or longitudinal welds and 300°F for circumferential welds at EOL (32 EFPY) and life extension (48 EFPY). Specifically, the intermediate to lower shell circumferential weld was the most limiting material with 32 and 48 EFPY PTS values of 116°F and 123°F respectively.

1 INTRODUCTION

A Pressurized Thermal Shock (PTS) Event is an event or transient in pressurized water reactors (PWRs) causing severe overcooling (thermal shock) concurrent with or followed by significant pressure in the reactor vessel. A PTS concern arises if one of these transients acts on the beltline region of a reactor vessel where a reduced fracture resistance exists because of neutron irradiation. Such an event may produce a flaw or cause the propagation of a flaw postulated to exist near the inner wall surface, thereby potentially affecting the integrity of the vessel.

The purpose of this report is to determine the RT_{PTS} values for the Byron Unit 2 reactor vessel using the fluence results of the 5% uprating and the previous PTS Report in WCAP-15177[10]. Section 2.0 discusses the PTS Rule and its requirements. Section 3.0 provides the methodology for calculating RT_{PTS}. Section 4.0 provides the reactor vessel beltline region material properties for the Byron Unit 2 reactor vessel. The neutron fluence values used in this analysis are presented in Section 5.0 and were obtained from SAE-REA-00-546[11]. The results of the RT_{PTS} calculations are presented in Section 6.0. The conclusion and references for the PTS evaluation follow in Sections 7.0 and 8.0, respectively.

2 PRESSURIZED THERMAL SHOCK RULE

The Nuclear Regulatory Commission (NRC) amended its regulations for light-water-cooled nuclear power plants to clarify several items related to the fracture toughness requirements for reactor pressure vessels, including pressurized thermal shock requirements. The latest revision of the PTS Rule[1], 10 CFR Part 50.61, was published in the Federal Register on December 19, 1995, with an effective date of January 18, 1996.

This amendment to the PTS Rule makes three changes:

1. The rule incorporates in total, and therefore makes binding by rule, the method for determining the reference temperature, RT_{NDT}, including treatment of the unirradiated RT_{NDT} value, the margin term, and the explicit definition of "credible" surveillance data, which is also described in Regulatory Guide 1.99, Revision 2[2].

2. The rule is restructured to improve clarity, with the requirements section giving only the requirements for the value for the reference temperature for end of license (EOL) fluence, RT_{PTS}.

3. Thermal annealing is identified as a method for mitigating the effects of neutron irradiation, thereby reducing RT_{PTS}.

The PTS Rule requirements consist of the following:

- For each pressurized water nuclear power reactor for which an operating license has been issued, the licensee shall have projected values of RT_{PTS}, accepted by the NRC, for each reactor vessel beltline material for the EOL fluence of the material.

- The assessment of RT_{PTS} must use the calculation procedures given in the PTS Rule, and must specify the bases for the projected value of RT_{PTS} for each beltline material. The report must specify the copper and nickel contents and the fluence values used in the calculation for each beltline material.

- This assessment must be updated whenever there is a significant change in projected values of RT_{PTS} or upon the request for a change in the expiration date for operation of the facility. Changes to RT_{PTS} values are significant if either the previous value or the current value, or both values, exceed the screening criterion prior to the expiration of the operating license, including any renewal term, if applicable for the plant.

- The RT_{PTS} screening criterion values for the beltline region are:

  270°F for plates, forgings and axial weld materials, and
  300°F for circumferential weld materials.

3 METHOD FOR CALCULATION OF RT_{PTS}

RT_{PTS} must be calculated for each vessel beltline material using a fluence value, f, which is the EOL fluence for the material. Equation 1 must be used to calculate values of RT_{NDT} for each weld and plate or forging in the reactor vessel beltline.

$$RT_{NDT} = RT_{NDT(U)} + M + \Delta RT_{NDT} \tag{1}$$

Where,

$RT_{NDT(U)}$ = Reference Temperature for a reactor vessel material in the pre-service or unirradiated condition

M = Margin to be added to account for uncertainties in the values of $RT_{NDT(U)}$, copper and nickel contents, fluence and calculational procedures. M is evaluated from Equation 2

$$M = \sqrt{\sigma_U^2 + \sigma_\Delta^2} \tag{2}$$

σ_U is the standard deviation for $RT_{NDT(U)}$.

σ_U = 0°F when $RT_{NDT(U)}$ is a measured value.

σ_U = 17°F when $RT_{NDT(U)}$ is a generic value.

σ_Δ is the standard deviation for RT_{NDT}.

For plates and forgings:

σ_Δ = 17°F when surveillance capsule data is not used.

σ_Δ = 8.5°F when surveillance capsule data is used.

For welds:

σ_Δ = 28°F when surveillance capsule data is not used.

σ_Δ = 14°F when surveillance capsule data is used.

σ_Δ not to exceed one half of ΔRT_{NDT}

ΔRT_{NDT} is the mean value of the transition temperature shift, or change in ΔRT_{NDT}, due to irradiation, and must be calculated using Equation 3.

$$\Delta RT_{NDT} = (CF) * f^{(0.28-0.10\log f)} \tag{3}$$

CF (°F) is the chemistry factor, which is a function of copper and nickel content. CF is determined from Tables 1 and 2 of the PTS Rule (10 CFR 50.61). Surveillance data deemed credible must be used to determine a material-specific value of CF. A material-specific value of CF is determined in Equation 5.

The EOL Fluence (f) is the calculated neutron fluence, in units of 10^{19} n/cm^2 (E > 1.0 MeV), at the clad-base-metal interface on the inside surface of the vessel at the location where the material in question receives the highest fluence. The EOL fluence is used in calculating RT_{PTS}.

Equation 4 must be used for determining RT_{PTS} using Equation 3 with EOL fluence values for determining RT_{PTS}..

$$RT_{PTS} = RT_{NDT(U)} + M + \Delta RT_{PTS} \tag{4}$$

To verify that RT_{NDT} for each vessel beltline material is a bounding value for the specific reactor vessel, licensees shall consider plant-specific information that could affect the level of embrittlement. This information includes but is not limited to the reactor vessel operating temperature and any related surveillance program results. Results from the plant-specific surveillance program must be integrated into the RT_{NDT} estimate if the plant-specific surveillance data has been deemed credible.

A material-specific value of CF for surveillance materials is determined from Equation 5.

$$CF = \frac{\sum[A_i * f_i^{(0.28-0.10\log f_i)}]}{\sum[f_i^{(0.56-0.20\log f_i)}]} \tag{5}$$

In Equation 5, "A_i" is the measured value of ΔRT_{NDT} and "f_i" is the fluence for each surveillance data point. If there is clear evidence that the copper and nickel content of the surveillance weld differs from the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of RT_{NDT} must be adjusted for differences in copper and nickel content by multiplying them by the ratio of the chemistry factor for the vessel material to that for the surveillance weld.

Irradiation temperature and fluence (or fluence factor) are first order environmental variables in assessing irradiation damage. To account for differences in temperature between surveillance specimens and vessel, an adjustment to the data must be performed. Studies have shown that for temperatures near 550°F, a 1°F decrease in irradiation temperature will result in approximately a 1°F increase in ΔRT_{NDT}. For capsules with irradiation temperature of $T_{capsule}$ and a plant with an irradiation temperature of T_{plant}, an adjustment to normalize $\Delta RT_{PTS, measured}$ to T_{plant} is made as follows:

$$\text{Temp. Adjusted } \Delta RT_{PTS} = \Delta RT_{PTS\ measured} + 1.0*(T_{capsule} - T_{plant})$$

Note that the temperature adjust methodology has been reinforce by the NRC at the NRC Industry Meetings on November 12, 1997 and February 12, 13 of 1998.

4 VERIFICATION OF PLANT SPECIFIC MATERIAL PROPERTIES

Before performing the pressurized thermal shock evaluation, a review of the latest plant-specific material properties for the Byron Unit 2 vessel was performed. The beltline region of a reactor vessel, per the PTS Rule, is defined as "the region of the reactor vessel (shell material including welds, heat affected zones and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage". Figure 1 identifies and indicates the location of all beltline region materials for the Byron Unit 2 reactor vessel.

The best estimate copper and nickel contents of the beltline materials were obtained from WCAP-15177[10]. The best estimate copper and nickel content is also documented in Table 1 herein. The average values were calculated using all of the available material chemistry information. Initial RT_{NDT} values for Byron Unit 2 reactor vessel beltline material properties are also shown in Table 1. As a note, per WCAP-15176[5], Weld WF-614 experiences less the 10^{17} n/cm^2 for both 32 and 48 EFPY

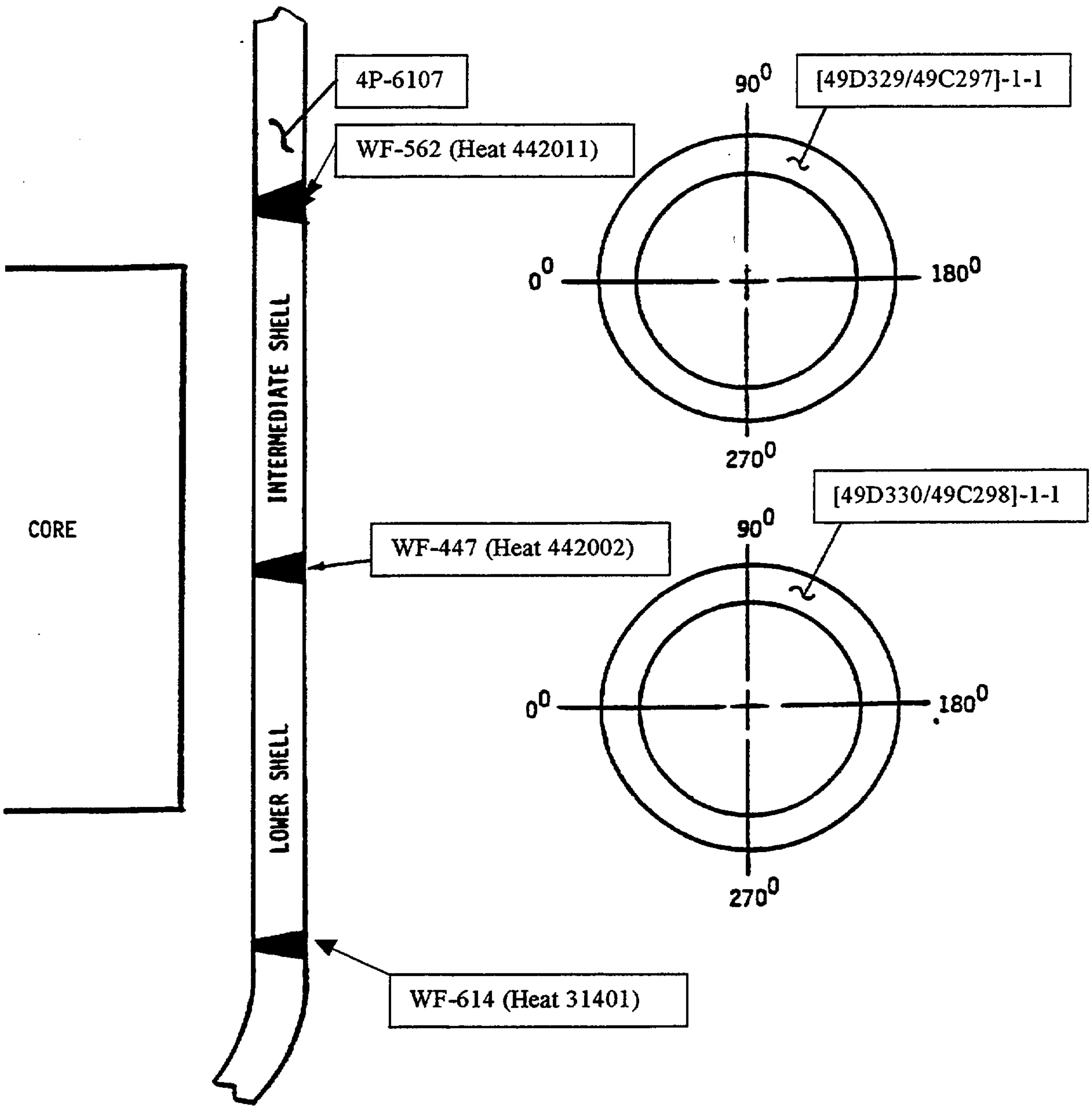


Figure 1: Identification and Location of Beltline Region Materials for the Byron Unit 2 Reactor Vessel

Table 1[b]

Byron Unit 2 Reactor Vessel Beltline Unirradiated Material Properties

Material Description	**Cu (%)**	**Ni(%)**	**Initial RT_{NDT}[a]**
Closure Head Flange 5P7382 / 3P6407	---	0.71	0
Vessel Flange 124L556VA1	---	0.70	30
Nozzle Shell Forging 4P-6107[b]	0.05	0.74	10
Intermediate Shell Forging [49D329/49C297]-1-1	0.01	0.70	-20
Lower Shell Forging [49D330/49C298]-1-1	0.06	0.73	-20
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat # 442002)[b]	0.04	0.63	10
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-562 (Heat # 442011)	0.03	0.67	40
Byron Unit 1 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.69	---
Byron Unit 2 Surveillance Program Weld Metal (Heat # 442002)	0.02	0.71	---
Braidwood Units 1 & 2 Surveillance Program Weld Metals (Heat # 442011)	0.03	0.67, 0.71	---

Notes:

(a) The initial RT_{NDT} values for the plates and welds are based on measured data.

(b) Table duplicated from Table 1 of Reference 10.

5 NEUTRON FLUENCE VALUES

The calculated fast neutron fluence (E > 1.0 MeV) values at the inner surface of the Byron Unit ~~1~~ 2 reactor vessel for 32 and 48 EFPY are shown in Table 2. These values were projected using the fluence results of the 5% uprating analysis[11].

TABLE 2
Fluence (E > 1.0 MeV) on the Pressure Vessel Clad/Base Interface for Byron Unit 2
at 32 (EOL) and 48 (Life Extension) EFPY

Material	Location	32 EFPY Fluence	48 EFPY Fluence
Nozzle Shell Forging 4P-6107	45°	5.22×10^{18} n/cm^2	7.84×10^{18} n/cm^2
Intermediate Shell Forging [49D329/49C297]-1-1	45°	2.06×10^{19} n/cm^2	3.10×10^{19} n/cm^2
Lower Shell Forging [9D330/49C298]-1-1	45°	2.06×10^{19} n/cm^2	3.10×10^{19} n/cm^2
Intermediate to Lower Shell Forging Circ. Weld Seam WF-447 (Heat 442002)	45°	2.03×10^{19} n/cm^2	3.05×10^{19} n/cm^2
Nozzle Shell to Intermediate Shell Forging Circ. Weld Seam WF-562 (Heat 442011)	45°	5.22×10^{18} n/cm^2	7.84×10^{18} n/cm^2

6 DETERMINATION OF RT_{PTS} VALUES FOR ALL BELTLINE REGION MATERIALS

Using the prescribed PTS Rule methodology, RT_{PTS} values were generated for all beltline region materials of the Byron Unit 2 reactor vessel for fluence values at the EOL (32 EFPY) and life extension (48 EFPY).

Each plant shall assess the RT_{PTS} values based on plant-specific surveillance capsule data. For Byron Unit 2, the related surveillance program results have been included in this PTS evaluation. (See Reference 8 for the credibility evaluation of the Byron Unit 2 surveillance data.)

As presented in Table 3, chemistry factor values for Byron Unit 2 based on average copper and nickel weight percent were calculated using Tables 1 and 2 from 10 CFR 50.61[1]. Additionally, chemistry factor values based on credible surveillance capsule data from Byron Units 1 and 2[8, 12], and Braidwood Units 1 and 2 are calculated in Table 4a and 4b, respectively. Tables 5 and 6 contain the RT_{PTS} calculations for all beltline region materials at EOL (32 EFPY) and life extension (48 EFPY).

TABLE 3

Interpolation of Chemistry Factors Using Tables 1 and 2 of 10 CFR Part 50.61

Material	**Ni, wt %**	**Chemistry Factor, °F**
Intermediate Shell Forging [49D329/49C297]-1-1 Given Cu wt% = 0.01	0.70	20.0°F
Lower Shell Forging [49D330/49C298]-1-1 Given Cu wt% = 0.06	0.73	37.0°F
Nozzle Shell Forging 4P-6107 Given Cu wt% = 0.05	0.74	31.0°F
Intermediate to Lower Shell Circ. Weld WF-447 Given Cu wt% = 0.04	0.63	54.0°F
Nozzle Shell to Intermediate Shell Circ. Weld WF-562 Given Cu wt% = 0.03	0.67	41.0°F
Byron Unit 1 and 2 Surveillance Program Weld Metal Given Cu wt% = 0.02	0.69, 0.71	27.0°F
Braidwood Unit 1 and 2 Surveillance Program Weld Metal Given Cu wt% = 0.03	0.67, 0.71	41.0°F

TABLE 4a[a]

Calculation of Chemistry Factors using Byron Unit 2 Surveillance Capsule Data

Material	Capsule	Capsule f	FF[b]	ΔRT_{NDT}[c]	$FF*\Delta RT_{NDT}$	FF^2
Lower Shell Forging [49D330/49C298]-1-1 (Tangential)	U	0.405	0.749	0.0[e]	0	0.561
	W	1.27	1.067	3.65	3.89	1.138
	X	2.30	1.225	15.75	19.29	1.500
Lower Shell Forging [49D330/ 49C298]-1-1	U	0.405	0.749	19.76	14.80	0.561
	W	1.27	1.067	31.88	34.02	1.138
	X	2.30	1.225	38.91	47.66	1.500
				SUM:	119.66	6.398
	$CF_{Forging} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (119.66) \div (6.398) =$ **18.7°F**					
Byron Unit 1 Surv. Weld Material (Heat # 442002)	U	0.404	0.749	11.22 (5.61)[d]	8.40	0.561
	X	1.57	1.125	80.22 (40.11)[d]	90.25	1.266
	W	2.43	1.239	102.68 (51.34)[d]	127.22	1.535
Byron Unit 2 Surv. Weld Material (Heat # 442002)	U	0.405	0.749	16.88 (8.44)[d]	12.64	0.561
	W	1.27	1.067	57.76 (28.88)[d]	61.63	1.138
	X	2.30	1.225	108.02 (54.01)[d]	132.32	1.500
				SUM:	432.46	6.561
	$CF_{Surv.\ Weld,\ 442002} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (432.46) \div (6.561) =$ **65.9°F**					

Notes:

(a) The calculation for the Byron Unit 2 chemistry factor was taken from Reference 13 & 14

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (See Ref. 13 & 14).

(d) The Byron 1 & 2 surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of 2.00. No temperature adjustment are required.

TABLE 4b

Calculation of Chemistry Factors using Braidwood Units 1 & 2 Surveillance Capsule Data

Material	Capsule	Capsule f	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Braidwood Unit 1	U	0.387	0.737	17.06[d]	12.57	0.543
Surveillance	X	1.24	1.060	30.15[d]	31.96	1.124
Weld Heat 442011, WF-501	W	2.09	1.201	49.68[d]	59.67	1.442
Braidwood Unit 2	U	0.400	0.746	0.0[d]	0	0.557
Surveillance	X	1.23	1.058	26.3[d]	27.83	1.119
Weld Heat 442011, WF-501	W	2.25	1.220	23.9[d]	29.16	1.488
				SUM:	161.19	6.273
	CF = $\Sigma(FF * RT_{NDT}) \div \Sigma(FF^2)$ = (161.19) ÷ (6.273) = **25.7°F**					

Notes:

(a) The calculation for the Braidwood Units 1 and 2 chemistry factor was taken from Reference 15.

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values (See Ref. 15)

(d) The Braidwood 1 & 2 surveillance weld metal ΔRT_{NDT} values do not require a ratio factor or temperature adjustment.

TABLE 5
RT_{PTS} Calculation for Byron Unit 2 Beltline Region Materials at EOL (32 EFPY)

Material	Fluence, x 10^{19} (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[c] (°F)	Margin (°F)	$RT_{NDT(U)}$[a] (°F)	RT_{PTS}[b] (°F)
Intermediate Shell Forging	2.06	1.20	20	24.0	24	-20	28
Lower Shell Forging	2.06	1.20	37	44.4	34	-20	58
Lower Shell Forging (Using S/C Data)	2.06	1.20	18.7	22.4	17	-20	19
Nozzle Shell Forging 4P-6107	0.522	0.818	31	25.4	25.4	10	61
Inter. to Lower Shell Circ. Weld Metal	2.03	1.19	54	64.3	56	10	130
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	2.03	1.19	65.9	78.4	28	10	116
Nozzle Shell to Inter. Shell Circ. Weld Metal	0.522	0.818	41	33.5	33.5	40	107
Nozzle Shell to Inter. Shell Circ. Weld Metal (Using S/C Data)	0.522	0.818	25.7	21.0	21.0	40	82

Notes:

(a) Initial RT_{NDT} values are measured values (See Table 1)

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)

(c) $\Delta RT_{PTS} = CF * FF$

TABLE 6

RT_{PTS} Calculation for Byron Unit 2 Beltline Region Materials at Life Extension (48 EFPY)

Material	Fluence, x 10^{19} (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[c] (°F)	Margin (°F)	$RT_{NDT(U)}$[a] (°F)	RT_{PTS}[b] (°F)
Intermediate Shell Forging	3.10	1.30	20	26.0	26.0	-20	32
Lower Shell Forging	3.10	1.30	37	48.1	34	-20	62
Lower Shell Forging (Using S/C Data)	3.10	1.30	18.7	24.3	17	-20	21
Nozzle Shell Forging 4P-6107	0.784	0.93	31	28.8	28.8	10	68
Inter. to Lower Shell Circ. Weld Metal	3.05	1.29	54	69.7	56	10	136
Inter. to Lower Shell Circ. Weld Metal → Using S/C Data	3.05	1.29	65.9	85.0	28	10	123
Nozzle Shell to Inter. Shell Circ. Weld Metal	0.784	0.93	41	38.1	38.1	40	116
Nozzle Shell to Inter. Shell Circ. Weld Metal (Using S/C Data)	0.784	0.93	25.7	23.9	23.9	40	88

Notes:

(a) Initial RT_{NDT} values are measured values (See Table 1)

(b) $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin}$ (°F)

(c) $\Delta RT_{PTS} = CF * FF$

7 CONCLUSIONS

As shown in Tables 5 and 6, all of the beltline region materials in the Byron Unit 2 reactor vessel have EOL (32 EFPY) RT_{PTS} and Life Extension (48 EFPY) RT_{PTS} values below the screening criteria values of 270°F for plates, forgings and longitudinal welds and 300°F for circumferential welds. Specifically, the intermediate to lower shell circumferential Weld, WF-447, was the most limiting material with 32 and 48 EFPY PTS values of 116°F and 123°F respectively.

8 REFERENCES

1 10 CFR Part 50.61, "Fracture Toughness Requirements For Protection Against Pressurized Thermal Shock Events", Federal Register, Volume 60, No. 243, dated December 19, 1995.

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

3 WCAP-10398, "Commonwealth Edison Company Byron Station Unit No. 2 Reactor Vessel Radiation Surveillance Program", L. R. Singer, December, 1983.

4 WCAP-14824, Revision 2, "Byron Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and Surveillance weld Metal Integration for Byron & Braidwood", T. J.Laubham, et. al., November, 1997.

5 WCAP-15176, "Analysis of Capsule X from the Commonwealth Edison Co. Byron Unit 2 Reactor Vessel Radiation Surveillance Program", T. J. Laubham, et al., March 1999.

6 Letters CAE-97-231, CCE-97-314, "ComEd Response to NRC Question to WCAP-14940, WCAP-14970 and 14824 Rev. 2", From C.S. Hauser to Mr. Guy DeBoo (of ComEd), Dated January 6, 1998.

7. ComEd Letter to U.S. Regulatory Commission, "Response to Additional Information Regarding Reactor Pressure Vessel", From R.M. Krich, Dated September 3, 1998.

8. WCAP-15180, "Commonwealth Edison Co. Byron Unit 2 Surveillance Program Credibility Evaluation", T. J. Laubham, et al., March 1999.

9. WCAP-15123, Revision 1, "Analysis of Capsule W from Commonwealth Edison Company Byron Unit 1 Reactor Vessel Radiation Surveillance Program", T.J. Laubham, et. al., January 1999.

10. WCAP-15177, "Evaluation of Pressurized Thermal Shock for Byron Unit 2", T.J. Laubham, June 1999.

11. SAE-REA-00-546, "Reactor Vessel Neutron Exposure Projections for the Byron/Braidwood Uprating", J.D. Perock, January 12, 2000.

12. WCAP-15183, "Commonwealth Edison Company Byron Unit 1 Surveillance Program Credibility Evaluation", T.J. Laubham, June 1999.

13. WCAP-15178, "Byron Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, June 1999.

14. WCAP-15124, "Byron Unit 1 Heatup and Cooldown for Normal Operation", T. J. Laubham, November 1998.

15. WCAP-15373, "Braidwood Unit 2 Heatup and Cooldown for Normal Operation", T. J. Laubham, March 2000.