April 11, 2017

MEMORANDUM TO:	John W. Lubinski, Director Division of Engineering Office of Nuclear Reactor Regulation
	Joseph G. Giitter, Director Division of Risk Assessment Office of Nuclear Reactor Regulation
	George A. Wilson, Director Division of License Renewal Office of Nuclear Reactor Regulation
FROM:	David L. Rudland, Branch Chief ***/RA/*** Vessels & Internals Integrity Branch Division of Engineering Office of Nuclear Reactor Regulation
SUBJECT:	CLOSURE MEMORANDUM SUPPORTING THE LIMITED REVISION OF NUREG-0800 BRANCH TECHNICAL POSITION 5-3, “FRACTURE TOUGHNESS REQUIREMENTS”

This memorandum with seven enclosures is a closure document concluding the U. S. Nuclear Commission’s (NRC’s) effort in the past three years in resolving the issue of non-conservatism in NUREG-0800 Branch Technical Position (BTP) 5-3, “Fracture Toughness Requirements.” Plants using the American Society of Mechanical Engineers Boiler and Pressure Vessel Code editions older than 1973 may not have complete test data to determine nil-ductility transition temperature (T_{NDT}), nil-ductility transition reference temperature (RT_{NDT}), and upper-shelf energy (USE). Consequently, BTP 5-3 was issued to provide alternative methods to estimate these material properties. However, the industry indicated by a letter dated January 30, 2014 (Agency Document Access and Management System (ADAMS) Accession No. ML14038A265) that using a specific position in BTP 5-3 may not be conservative in predicting the initial RT_{NDT}. This memorandum summarized the NRC’s reassessment of all positions in BTP 5-3, including assessment of the industry’s two summary reports on BTP 5-3 non-conservatism.

In order to make this assessment, the staff considered both deterministic and probabilistic analyses in making the safety determination. Analyses described in Enclosures 1 to 4 are based on a deterministic approach of investigating the margin needed on the RT_{NDT} to account for the BTP5-3 non-conservatism. However, the staff determined that the margins determined in the early enclosures will not be implemented in the revised BTP 5-3 for 72 effective full power years (EFPYs) due to the results of the risk-informed analyses described in Enclosures 5 and 6. These analyses suggest that the change in risk associated with the BTP5-3 non-conservatism is

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sufficiently small and does not meet the NRC criterion for implementing a new requirement. A brief discussion of the results from each enclosure is given below.

Enclosure 1 documents the reassessment of non-conservatism in BTP 5-3 Positions B1.1(1) and B1.1(2) regarding the T_{NDT} estimation. Considering that the initial RT_{NDT} value of a material is determined by both the T_{NDT} estimation and the Charpy V-Notch (CVN) curve estimation, the NRC staff performed an error analysis to determine the interaction between the uncertainties of the T_{NDT} estimation and the uncertainties of the CVN curve estimation. These uncertainties were generated by the Office of Nuclear Regulatory Research (RES) for a variety of unirradiated nuclear-grade steel plates and forgings that were reported to the NRC as part of 10 CFR 50 Appendix H surveillance programs. The NRC staff's error analysis described in this enclosure suggests that considering additional margin on the RT_{NDT} is an acceptable way to ensure conservative behavior when using T_{NDT} from these positions in pressure-temperature (P-T) limits and pressurized thermal shock (PTS) evaluations.

Enclosure 2 documents the NRC staff's technical assessment of MRP-401 (BWRVIP-287), "Assessment of the Use of NUREG-0800 Branch Technical Position 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels" (ADAMS Accession No. ML15265A040), which supports the industry's opinion that there is no need to consider BTP 5-3 non-conservatism in P-T limits and PTS evaluations for pressurized water reactors (PWRs) for 60 years of operation. This enclosure focuses on the reassessment of non-conservatism in BTP 5-3 Positions B1.1(3) and B1.1(4) regarding the RT_{NDT} estimation and B1.2 regarding the USE estimation. For Position B1.1(3), the NRC staff verified the MRP's statistical analyses and concludes that considering additional margin is acceptable to ensure conservative behavior when using RT_{NDT} from this position in P-T limits and PTS evaluations. For Positions B1.1(4), since the MRP report's information is not sufficient to support a statistical analysis, the NRC staff performed a modified statistical analysis based on the RES available data and concludes that considering a similar additional margin is acceptable to ensure conservative behavior in P-T limits and PTS evaluations. For Position 1.2, the NRC staff examined the RES data and determined that the current practice is acceptable. The chapter on P-T limits based on probabilistic fracture mechanics (PFM) analysis in MRP-401 was not reviewed because the NRC staff relied on more comprehensive risk-informed P-T limits analyses in Enclosure 6 to make the regulatory decision.

Enclosure 3 documents the NRC staff's technical assessment of PWROG-15003-NP, Revision 0, "Material-Orientation Toughness Assessment (MOTA) for the Purposes of Mitigating Branch Technical Position (BTP) 5-3 Uncertainties" (ADAMS Accession No. ML15268A086), which supports the industry's opinion that there is no need to consider BTP 5-3 non-conservatism in the current P-T limits for PWRs. The NRC staff examined the definition and calculations of the MOTA margin in the report, which results from consideration of the flaw orientation and the fracture toughness orientation. The NRC staff also performed supplemental calculations to amend a case where the reported MOTA margin is insufficient to cover all PWRs with the BTP 5-3 forgings. Based on the evaluation, the NRC staff concludes that the deterministic MOTA margin approach is technically appropriate to support the current licensing basis (CLB) P-T limits in light of the BTP 5-3 non-conservatism.

Enclosure 4 documents the NRC staff's plant-specific assessment of the PWRs with the BTP 5-3 plates or forgings. The NRC staff performed a deterministic analysis by comparing the margin between the RT_{NDT} for the limiting material of the CLB P-T limits and the CLB RT_{NDT} for the material based on BTP 5-3. The analyses suggest that the combined plant-specific and generic safety margins in each PWR RPV bound the non-conservatism in BTP 5-3, and thus the staff concludes that there is no need to consider BTP 5-3 non-conservatism in P-T limits and PTS evaluations for PWRs for 60 years of operation.

Enclosure 5 documents the NRC staff's risk-informed assessment of the PTS evaluations due to BTP 5-3 non-conservatism. The NRC staff examined the documented PFM results in NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," 2007 (ADAMS Accession No. ML070860156) and found that instead of conducting PFM analyses, the risk values can be obtained from the plots of the risk value versus the reference temperature in this NUREG. Consequently, the NRC staff used the shifted reference temperature to account for the BTP 5-3 non-conservatism and generated change-in-risk values for the bounding PWRs for 72 EFPYs. Considering the insignificant change-in-risk values with and without the BTP 5-3 non-conservatism, the NRC staff concludes that there is no need to consider BTP 5-3 non-conservatism in PTS evaluations for PWRs for 72 EFPYs.

Enclosure 6 documents the NRC staff's risk-informed assessment of the P-T limits due to BTP 5-3 non-conservatism. Since documented PFM results on P-T limits do not exist, the NRC staff relied on PFM analyses performed by RES to support this effort. RES's PFM analyses account for the BTP 5-3 non-conservatism by increasing the uncertainty in the initial RT_{NDT} of the BTP 5-3 material using a specific standard deviation as an input to the PFM simulations. Considering the insignificant change-in-risk values with and without the BTP 5-3 non-conservatism, the NRC staff concludes that there is no need to consider BTP 5-3 non-conservatism in P-T limits evaluations for PWRs and BWRs for 72 EFPYs.

Enclosure 7 is the proposed BTP 5-3 with a new paragraph directing readers to this NRC memorandum, which concludes that there is no impact to P-T limits evaluations for PWRs and BWRs and PTS evaluations for PWRs for 72 EFPYs.

Based on the seven enclosures, the Vessels and Internals Integrity Branch concludes that the limited BTP 5-3 revision in Enclosure 7 is based on sound analytical and statistical evaluation results, considering all available material test data from many sources, including the domestic fleet surveillance data. Specifically, the deterministic analyses documented in Enclosures 1 to 4 indicate that considering additional margin is acceptable to ensure conservative behavior when using RT_{NDT} from the BTP 5-3 positions in deterministic P-T limits and PTS evaluations. However, these additional margins will not be implemented in the revised BTP 5-3 for 72 EFPYs because the risk-informed analyses in Enclosures 5 and 6 indicate that the small change in risk due to the BTP 5-3 non-conservatism does not meet the NRC criterion for implementing a new requirement. The conclusion drawn in this memorandum is consistent with the guidance in NUREG/BR-0058, "Regulatory Analysis Guideline of the U. S. Nuclear Regulatory Commission." Accordingly, there is no need to consider the BTP 5-3 non-conservatism in P-T limits evaluations for PWRs and BWRs and PTS evaluations for PWRs for 72 EFPYs.

Enclosures:
1. Reexamination of BTP 5-3 Positions B1.1(1) and B1.1(2)
2. Technical Assessment of MRP-401
3. Technical Assessment PWROG-15003-NP, Revision 0
4. Plant-Specific Review of PWRs with BTP 5-3 Plates or Forgings
5. Risk Assessment of Operation under Pressurized Thermal Shock Event for 72 EFPY
6. Risk Assessment of Operation under Pressure-Temperature Limits for 72 EFPY
7. The Proposed BTP 5-3 Revision

CLOSURE MEMORANDUM SUPPORTING THE LIMITED REVISION OF NUREG-0800 BRANCH TECHNICAL POSITION 5-3, “FRACTURE TOUGHNESS REQUIREMENTS” – DATE: April 11, 2017

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<u>ENCLOSURE 1</u>

TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING ON REEXAMINATION OF BTP 5-3 POSITIONS B1.1(1) AND B1.1(2) REGARDING T_{NDT} ESTIMATION USING DOMESTIC FLEET DATA

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TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING ON REEXAMINATION OF BTP 5-3 POSITIONS B1.1(1) AND B1.1(2) REGARDING T_{NDT} ESTIMATION USING DOMESTIC FLEET DATA

1.0 INTRODUCTION AND BACKGROUND

In response to AREVA's January 30, 2014, letter asserting that using Position B1.1(4) of NUREG-0800 Branch Technical Position (BTP) 5-3 may be non-conservative (Agency Document Access and Management System (ADAMS) Accession No. ML14038A265), the U.S. Nuclear Regulatory Commission (NRC) staff has reviewed all positions in BTP 5-3 to ensure that any non-conservatisms in other positions of BTP 5-3, if exist, have also been evaluated. Similarly, industry has performed an evaluation of this issue and produced two reports addressing the non-conservatism in applying BTP 5-3 positions: (1) MRP-401 (BWRVIP-287), "Assessment of the Use of NUREG-0800 Branch Technical Position 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels," September 2015 (ADAMS Accession No. ML15265A040) and (2) Pressurized-Water Reactor Owners Group (PWROG)-15003-NP, Revision 0, "Material-Orientation Toughness Assessment (MOTA) for the Purposes of Mitigating Branch Technical Position (BTP) 5-3 Uncertainties," June 2015 (ADAMS Accession No. ML15268A086). Unfortunately, neither of these reports evaluated BTP 5-3 Positions B1.1(1) and B1.1(2) regarding estimation of the nil-ductility transition temperature (T_{NDT}). The NRC staff determined that such an evaluation should be performed. This assessment documents the NRC staff's reexamination of BTP 5-3 Positions B1.1(1) and B1.1(2) regarding T_{NDT} determination.

2.0 REGULATORY ASSESSMENT

Appendix G, "Fracture Toughness Requirements," to Title 10 of *Code of Federal Regulations* Part 50 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (RCPB) of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation. Fracture toughness of the RCPB ferritic materials, including the reactor pressure vessel, is evaluated using the nil-ductility transition reference temperature (RT_{NDT}) and the upper-shelf energy (USE). RT_{NDT} is determined by T_{NDT} from the drop-weight test (in accordance with ASTM International Standard E 208, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels") and Charpy V-notch (CVN) impact tests in accordance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, NB-2331, "Material for Vessels." Mathematically, NB-2331 defines RT_{NDT} as the greater of T_{NDT} and [$T_{(T)50}$ – 60 °F], where $T_{(T)50}$ is the temperature of the CVN curve (based on transverse CVN specimens) corresponding to 50 ft-lb of impact energy. Determination of USE will not be presented here because it is not the subject of the current assessment.

ENCLOSURE 1

Since plants using the ASME Code editions older than 1973 may not have complete test data to determine T_{NDT}, RT_{NDT}, or USE, BTP 5-3 was issued to provide alternative methods to estimate these parameters: Positions B1.1(1) and B1.1(2) are for T_{NDT}; Positions B1.1(3)a, B1.1(3)b, and B1.1(4) are for RT_{NDT}; and Position 1.2 is for USE. Position B1.1(1) provides a procedure to estimate T_{NDT} for SA533 Grade B, Class 1 plates, and Position B1.1(2) provides a procedure to estimate T_{NDT} for SA-508 Class II forgings. For these older plants, BTP 5-3 has been used in the development of pressure-temperature (P-T) limits, pressurized thermal shock (PTS) evaluations, and flaw evaluations.

3.0 TECHNICAL ASSESSMENT

To assist in this work, the NRC Office of Nuclear Regulatory Research (RES) developed a technical letter report, TLR-RES/DE/CIB-2014-011, December 9, 2016 (NRC TLR; ADAMS Accession No. ML16341B108), "Assessment of Predictions of RT_{NDT} and Upper Shelf Energy Made Using Branch Technical Position 5-3." This NRC TLR describes a database of information on unirradiated nuclear-grade steel plates and forgings reported to the NRC as part of 10 CFR 50 Appendix H surveillance programs, which were selected because they are available in the public domain, they are directly representative of the U.S. operating fleet, and they have both raw T_{NDT} and CVN data in both the transverse and longitudinal orientations. For this assessment, the full longitudinal CVN curves developed by the RES in support of the NRC TLR are used to estimate T_{NDT} values in accordance with Positions B1.1(1) and B1.1(2), and the test-based T_{NDT} values compiled by the RES in support of the NRC TLR are used as the base values for comparison.

Position B1.1(1)

BTP 5-3 Position B1.1(1) specifies, for SA-533 Grade B, Class 1 plate and weld material, that when a full longitudinal CVN curve is available but T_{NDT} is not, T_{NDT} may be estimated as the higher of the temperature of the longitudinal CVN curve corresponding to 30 ft-lbs or 0 °F. The data analysis results related to Position B1.1(1) are presented in Figure 7.1 of the NRC TLR. Figure 7.1 shows that three data points (out of a total of 31), having T_{NDT} of 10 °F established by the ASTM International Standard E 208 procedure, are not bounded by Position B1.1(1). From Table 6-1 of the NRC TLR, the NRC staff confirmed that the three points could be Plates C5114-1, C4487-1, C4489-1, C4441-1, C5286-1, or C4007-1. It should be noted that these three data points exceed the BTP 5-3 line by a small amount (10 °F to the upper bound line), and the distance between the BTP 5-3 line and the lower bound line is 65 °F. Therefore, if T_{NDT} is controlling (i.e., T_{NDT} is the initial RT_{NDT}), the associated error of non-conservatism is 10 °F. On the other hand, if the CVN curve is controlling (i.e., [$T_{(T)50}$ – 60 °F] based on the CVN curve is the initial RT_{NDT}), the associated error to the upper and lower bound line is about 40 °F (i.e., a 2σ value about 40 °F) based on Figure 4-5 of MRP-401 for all available plates using BTP 5-3 Position B1.1(3). The following will determine the correct σ to be used as the standard deviation for initial RT_{NDT} (σ_I) in the Margin calculation, as defined in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," when T_{NDT} and [$T_{(T)50}$ – 60 °F] are close.

Figure 1 illustrates an assumed case where T_{NDT} based on Position B1.1(1) is 10 °F, and [$T_{(T)50}$ – 60 °F] based on Position B1.1(3)a or B1.1(3)b is 5 °F. Since T_{NDT} is higher, the initial RT_{NDT} is T_{NDT}, i.e., 10 °F. Figure 1 marked a yellow region on the top line, representing a generic error of 10 °F on the right hand side and a generic error of 65°F on the left hand side of T_{NDT}. The actual T_{NDT} for this material per drop-weight test could be any value within the yellow region of the upper figure, which covers approximately 95% of all probable T_{NDT} values. Similarly, the actual [$T_{(T)50}$ – 60 °F] for this material per CVN test could be any value within the yellow region (2σ is 40 °F) of the lower figure. The green line marked the region for the possible initial RT_{NDT} values that could be missed if a 2σ of 10 °F for T_{NDT} was used as the $2\sigma_I$ for the initial RT_{NDT}. Please note that using the error to the upper bound line as the 2σ associated with a normal distribution to represent a skewed distribution is conservative in this case where only the upper bound line matters. To cover the possibility that [$T_{(T)50}$ – 60 °F] is controlling (see Figure 1), a $2\sigma_I$ of at least 35 °F is needed. Therefore, when T_{NDT} based on Position B1.1(1) is used with [$T_{(T)50}$ – 60 °F] based on longitudinal CVN curve to determine the initial RT_{NDT} value for an SA-533 Grade B, Class 1 plate, no revision is needed for Position B1.1(1) as long as a σ_I of 20 °F is considered in the margin calculation. Enclosures 5 and 6 based on risk-informed approach further demonstrate that even this σ_I of 20 °F does not need to be considered in P-T limit and PTS evaluations for 72 effective full power years (EFPYs).

T_{NDT}

```
-----•--+--•-----•-----•-----•-----•-----•-----x-----+-----•-----•-----•-----
      -55°F                            0°F  10°F  20°F
```

[$T_{(T)50}$ – 60°F]

```
---------•-----•--+--•-----•-----•-----•--x--•-----•-----•-----•--+--•-----
               -35°F                     5°F    20°F          45°F
```

Upper yellow region: Possible value for T_{NDT}
Lower yellow region: Possible value for [$T_{(T)50}$ – 60 °F]
Green line: Possible value for RT_{NDT}

Figure 1 Possible values of T_{NDT}, [$T_{(T)50}$ – 60 °F], and initial RT_{NDT}

<u>Position B1.1(2)</u>

BTP 5-3 Position B1.1(2) specifies, for SA-508, Class II forgings, that when a full longitudinal CVN curve is available but T_{NDT} is not, T_{NDT} may be estimated as the lowest of (a) 60 °F, (b) the temperature of USE, and (c) the temperature for 100 ft-lb if USE is greater than 100 ft-lb. The data analysis results related to Position B1.1(2) are presented in Figure 8.1 of the NRC TLR. Figure 8.1 shows that only one data point is significantly not bounded by Position B1.1(2) and this data point (forging) has an official T_{NDT} of 40 °F. Table 6-1 confirmed that this point is Forging 5P-5933 (Byron 1).

A close examination of the test data of this forging indicated that the difference between the two candidates for the initial RT_{NDT}, the official T_{NDT} (40 °F) and the official [$T_{(T)50}$ – 60 °F] (- 94°F), is 134°F. For all other data points in Figure 8.1, the corresponding value is below 100 °F. Therefore, although Forging 5P-5933 may not be considered as an outlier statistically, it has unusual material characteristics, indicating that the test-based T_{NDT} of 40 °F may be incorrect (i.e., the T_{NDT} may be much lower than 40 °F). Regardless of this concern, the Forging 5P-5933 data are bounded by the upper $2\sigma_I$ line. Therefore, either T_{NDT} or [$T_{(T)50}$ – 60 °F] is controlling, no revision is needed for Position B1.1(2) as long as a σ_I of 20 °F is considered in the margin calculation. Enclosures 5 and 6 based on risk-informed approach further demonstrate that even this σ_I of 20 °F does not need to be considered in P-T limit and PTS evaluations for 72 EFPYs.

The above assessment of Positions B1.1(1) and B1.1(2) are based on a statistical analysis of the test data reported in the NRC TLR. This assessment shows that using the above σ_I values in the deterministic margin calculation for P-T limits and PTS evaluations is conservative. However, this approach will not be implemented in the revised BTP 5-3 for 72 EFPYs because the risk-informed approaches in Enclosures 5 and 6 indicated that the resulting changes in risk considering these σ_I values do not meet the NRC criterion for implementing a new requirement.

4.0 CONCLUSION

Based on the technical assessment discussed in Section 3.0, the NRC staff concludes that when T_{NDT} based on Position B1.1(1) for an SA-533 Grade B, Class 1 plate or Position B1.1(2) for an SA-508 Class II forging is used with a longitudinal CVN curve to determine the T_{NDT} value, an acceptable approach is to use a σ_I of 20 °F in the margin calculation if this T_{NDT} is determined to be the RT_{NDT}. However, use of this σ_I in the margin calculation for P-T limits and PTS evaluations will not be implemented in the revised BTP 5-3 for 72 EFPYs because the risk-informed approaches in Enclosures 5 and 6 indicated that the resulting changes in risk considering these σ_I values do not meet the NRC criterion for implementing a new requirement.

ENCLOSURE 2

TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING ON MRP-401 (BWRVIP-287), "ASSESSMENT OF THE USE OF NUREG-0800 BRANCH TECHNICAL POSITION 5-3 ESTIMATION METHODS FOR INITIAL FRACTURE TOUGHNESS PROPERTIES OF REACTOR PRESSURE VESSEL STEELS," REGARDING BTP 5-3 POSITIONS B1.1(3)A AND B1.1(3)B FOR RT_{NDT} ESTIMATION AND POSITION 1.2 FOR UPPER SHELF ENERGY

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TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING ON ELECTRIC POWER RESEARCH INSTITUTE REPORT MRP-401 (BWRVIP-287) "ASSESSMENT OF THE USE OF NUREG-0800 BRANCH TECHNICAL POSITION 5-3 ESTIMATION METHODS FOR INITIAL FRACTURE TOUGHNESS PROPERTIES OF REACTOR PRESSURE VESSEL STEELS," REGARDING BTP 5-3 POSITIONS B1.1(3)a and B1.1(3)b FOR RT_{NDT} ESTIMATION AND POSITION 1.2 FOR UPPER SHELF ENERGY

1.0 INTRODUCTION AND BACKGROUND

By E-mail dated September 23, 2015, Rory Swezey of the Electric Power Research Institute (EPRI) sent to Robert Hardies, U.S. Nuclear Regulatory Commission (NRC), MRP-401 (BWRVIP-287), "Assessment of the Use of NUREG-0800 Branch Technical Position [BTP] 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels" (Ref. 1) to assist the NRC's on-going effort related to the NUREG-0800 BTP 5-3 (Ref. 2) reassessment. This E-mail with the attached MRP-401 was sent in reply to the NRC's letter dated June 1, 2015 (Ref. 3) to EPRI, requesting the report for information.

NRC took action to reassess the BTP after industry and NRC's independent evaluations confirmed AREVA's January 30, 2014, assertion (Ref. 4) that using Position B1.1(4) of BTP 5-3 may be non-conservative. The NRC's evaluation is documented in Technical Letter Report (TLR)-RES/DE/CIB-2014-011 (NRC TLR), "Assessment of Predictions of RT_{NDT} and Upper Shelf Energy Made Using Branch Technical Position 5-3" (Ref. 5). The industry's evaluations are documented in MRP-401 mentioned above and PWROG-15003-NP, Revision 0, "Material-Orientation Toughness Assessment (MOTA) for the Purposes of Mitigating Branch Technical Position (BTP) 5-3 Uncertainties" (Ref. 6). Assessment of MRP-401 and consideration of its technical contents along with information from NRC's independent evaluations and other industry reports on this subject are essential to the BTP 5-3 reassessment. The purpose of Enclosure 2 is to document NRC's technical assessment of MRP-401. The NRC's technical assessment of PWROG-15003-NP, Revision 0 is documented in Enclosure 3.

2.0 REGULATORY ASSESSMENT

Appendix G, "Fracture Toughness Requirements," to Title 10 of *Code of Federal Regulations* (10 CFR) Part 50 (Ref. 7) specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (RCPB) of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation. The two most important material properties mentioned in Appendix G which affect fracture toughness of the RCPB ferrictic materials, including the reactor pressure vessel (RPV), are nil-ductility transition reference temperature (RT_{NDT}) and upper-shelf energy (USE). They are determined in accordance with American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, NB-2331 (Ref. 8) and ASTM International Standard (ASTM) E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels" (Ref. 9). Determination of initial (i.e., unirradiated) RT_{NDT} also requires information on nil-ductility transition temperature (T_{NDT}), which is from the drop weight test per ASTM E 208, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels" (Ref. 10).

Since plants using the ASME Code editions older than 1973 may not have complete test data to determine T_{NDT}, RT_{NDT}, or USE, BTP 5-3 was issued to provide alternative methods to determine them: Positions B1.1(1) and B1.1(2) for T_{NDT}; Positions B1.1(3)a, B1.1(3)b, and B1.1(4) for RT_{NDT}; and Position 1.2 for USE. For these older plants, BTP 5-3 has been used in determination of pressure-temperature (P-T) limits, pressurized thermal shock (PTS) evaluations, and flaw evaluations. It should be noted that the NRC TLR, MRP-401, and PWROG-15003-NP, Revision 0 all went beyond Position B1.1(4) of BTP 5-3 to include assessment of other positions in BTP 5-3. In addition, the NRC engaged industry in public meetings on June 4, 2014, February 19 and June 2, 2015, and January 19, 2016, to exchange information regarding resolution of this issue.

3.0 <u>TECHNICAL ASSESSMENT</u>

This assessment focuses only on the technical contents of MRP-401 supporting the NRC staff's reassessment of BTP 5-3. Therefore, the NRC staff skipped Section 1, "Introduction and Purpose," Section 2, "Background," Section 3, "Use of Branch Technical Position 5-3 in the U.S. Fleet," and Section 5, "Reactor Pressure Vessel Pressure-Temperature Limits." Section 5 is based on probabilistic fracture mechanics (PFM) evaluations for PWRs and is not reviewed because the NRC staff's review of a deterministic evaluation in PWROG-15003-NP, Revision 0, (see Enclosure 3) already concluded that the current P-T limits for PWRs do not need to be updated due to non-conservatism in BTP 5-3. There is also no need for the NRC staff to comment on Section 8, "Summary and Conclusions," because the NRC staff will rely on its own assessments and conclusions summarized in Sections 3.1 to 3.4 to guide the BTP 5-3 resolution. As such, this assessment shall not be considered as NRC endorsement of MRP-401.

Further, unlike other NRC safety evaluations which used requests for additional information as a vehicle to resolve issues of technical concerns, this assessment relies heavily on the NRC TLR and the NRC staff's own analyses and engineering calculations to resolve all issues.

3.1 <u>Assessment of Section 4, "Database Development and Assessment for Key Branch Technical Position 5-3 Paragraphs"</u>

Section 4 presents the industry's statistical evaluation of the discrepancies between the estimated transverse Charpy V-notch (CVN) temperature at 50 ft-lb based on BTP 5-3 Position B1.1(3)a (i.e., using the longitudinal CVN data) and the corresponding temperature based on the ASME Code (i.e., using the transverse CVN data). The same type of statistical evaluation is repeated for BTP 5-3 Position B1.1(3)b. To support such analyses, the database comprises only material data having complete CVN curves for both transverse and longitudinal orientations. Since the statistical data analysis in MRP-401 is similar to that in the NRC TLR, on which the Office of Nuclear Reactor Regulation (NRR) staff has reviewed (Ref. 11), the NRC staff used information in the NRC TLR to establish credibility of the information in this section of MRP-401 regarding database, statistical analyses, and results.

Regarding data sources, the NRC staff found that the database of this section is from five sources: (a) the industry surveillance material data in Radiation Embrittlement Archive Project (REAP), (b) EPRI Nuclear Pressure Vessel Steel Data Base (NP-933), (c) the industry non-surveillance material data collected by EPRI in 2014, (d) data from the paper by Smith & Ayres, and (e) data from IAEA-TECDOC-1230. For references of these five sources, please see Reference 1. In contrast, the NRC TLR's database is only from REAP, and of a size about a little less than half of the MRP-401 database. Therefore, the data analysis (error analysis) results presented in Section 4 of MRP-401 are more comprehensive and more likely to be free of bias than those based on a smaller database.

3.1.1 Positions B1.1(3)a and B1.1(3)b

Both the NRC TLR and MRP-401 employed the hyperbolic tangent curve fitting with the identical lower shelf energy for the CVN data. The NRC staff examined Figure 9-1 (plate + forging, plate, and forging) in the NRC TLR and Figures 4-1 (plate) and 4-2 (forging) in MRP-401 for the Position B1.1(3)a application and concludes:

- Both the NRC and the industry results are credible because they show consistent data patterns and values.
- Estimating the mean of CVN 50 ft-lb temperature per Position B1.1(3)a is conservative for plates, but slightly non-conservative for forgings.

Similarly, the NRC staff examined Figure 9-3 (plate + forging, plate, and forging) in the NRC TLR and Figures 4-3 (plate) and 4-4 (forging) in MRP-401 for the Position B1.1(3)b application and concludes:

- Both the NRC and the industry results are credible because they show consistent data patterns and values. However, it should be noted that the NRC TLR plotted in the horizontal axis of Figure 9-3 the longitudinal CVN 50 ft-lb temperature of the CVN curve, whereas the MRP-401 plotted in the horizontal axes of Figures 4-3 and 4-4 the longitudinal CVN 50 ft-lb temperature + 20 °F. To make an appropriate comparison, the x-values of the data in NRC Figure 9-3 have to be adjusted by adding 20 °F.
- Estimating the mean of CVN 50 ft-lb temperature per Position B1.1(3)b is slightly non-conservative for plates, but significantly non-conservative for forgings.

MRP-401 further combined Figure 4-1 and Figure 4-3 into Figure 4-5, which represents the error analysis results when Position B1.1(3)a and B1.1(3)b are applied for plates. Similarly, MRP-401 combined Figure 4-2 and Figure 4-4 into Figure 4-6 (or Figure 4-7 with Rotterdam forgings being identified), which represents the error analysis results when Positions B1.1(3)a and B1.1(3)b are applied for forgings. In the BTP reassessment, the NRC staff relies predominantly on MRP-401 Figure 4-5, Figure 4-7, and Table 4-1 in reassessing Position B1.1(3), considering correctness, compatibility with Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials” (Ref. 12), philosophical consistency with the existing BTP position if the previous two are maintained, simplicity, and easy implementation.

An Acceptable Conservative Approach

The NRC staff noted that Figure 4-5 showed balanced distribution of data on either side of the diagonal line (the line based on BTP 5-3 Positions B1.1(3)a and B1.1(3)b) for plates and Figure 4-7 showed conservative distribution of data around the diagonal line for non-Rotterdam forgings. Based on this observation, the NRC staff concludes that continued use of Position B1.1(3) to determine the mean initial RT_{NDT} values for plates and non-Rotterdam forgings is adequate. However, the error distribution of Figures 4-5 and 4-7 suggested that the existing practice of setting the standard deviation for the initial RT_{NDT} (σ_I) to zero for plates and forgings having only plant-specific longitudinal Charpy V-notch curve may be non-conservative. When Position B1.1(3) is used, one acceptable approach is to consider the σ_I value that reflects the data scatter of the generic error analysis of Section 4 of MRP-401. This approach is in accordance with the guidance in RG 1.99, Revision 2 regarding use of generic RT_{NDT} values:

> [G]eneric 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.
>
> * The class for estimating initial RT_{NDT} is generally determined, for the welds with which this guide is concerned, by the type of welding flux (Linde 80 or other); for base metal, by the ASTM Standard Specification.

Therefore, for applications using Position B1.1(3), one acceptable conservative approach is to use a σ_I of 20 °F for all plates and non-Rotterdam forgings and σ_I of 60 °F for Rotterdam forgings.

These σ_I values are based on Table 4-1 and shall be combined with the standard deviation for ΔRT_{NDT} (σ_Δ) in accordance with RG 1.99, Revision 2 to calculate the margin in estimating the adjusted reference temperature (ART) for irradiated materials:

$$\text{Margin} = 2\,(\sigma_I^2 + \sigma_\Delta^2)^{1/2}$$

For unirradiated materials, the margin reduces to $2\sigma_I$, and the estimation of the initial RT_{NDT} using Position B1.1(3) becomes a conventional approach of mean plus 2 times the standard deviation (mean + 2σ).

An Acceptable Alternative

A variation of this acceptable conservative approach is to use the information in Table 4-2 of MRP-401 or the similar approach in the NRC TLR. Both of the MRP-401 and the NRC TLR alternatives rely on use of the best-fit line (instead of the BTP line) and its associated σ_I for different groups of materials using Position B1.1(3)a, Position B1.1(3)b, or Position B1.1(3)a,b (i.e., combined the error analysis of both positions into one). Table 4-2 of MRP-401 provides 9 pairs of (mean, σ_I), and Table 13-2 of the NRC TLR provides 4. Among them, the NRC staff only verified the second and the fifth pairs of Table 4-2 of MRP-401 and Equation (10) of the NRC TLR. Based on the alternative approach, MRP-401 performed a PTS evaluation in Section 6 for the PWR RPV having a BTP 5-3 plate with the highest RT_{PTS} plate.

Both the acceptable conservative approach and the acceptable alternative with specific σ_I values are based on deterministic analyses. However, use of them in P-T limits and PTS evaluations will not be implemented in the revised BTP 5-3 for 72 effective full power years (EFPYs) because the risk-informed approaches in Enclosures 5 and 6 indicated that the resulting changes in risk considering these σ_I values do not meet the NRC criterion for implementing a new requirement.

3.1.2 Positions B1.1(4)

This MRP-401 section presents a probabilistic distribution of measured RT_{NDT} in Figure 4-10 for SA533B-1 plates and a distribution in Figure 4-11 for SA508-2 forgings. This information may be valuable in its own right, but is not useful in assessing the potential non-conservatism associated with this position. To assess the potential non-conservatism, in additional to rely on information in the NRC TLR, the NRC staff also performed a separate error analysis on thirteen A 508-2 and five A 508-3 forging materials from REAP by an approach based on the CVN data at a single temperature (i.e., following Position B1.1(4) faithfully). The results from the error analysis summarized below in Table 1 are significantly different from those in Figure 10-4 of the NRC TLR for the identical materials by an approach based on the CVN curve (analytically oriented, but deviating from Position B1.1(4)). Consequently, the forging data in Figure 10-4 of the NRC TLR will not be relied on for decision making because the approach did not follow Position B1.1(4) faithfully and is overly conservative as discussed below.

Table 1 Error between RT_{NDT} Estimates by BTP 5-3 Position B1.1(4) and by ASME NB-2331

No.	Heat	Plant	T_{NDT} (°F)	T/CVN E (L) (°F/ft-lb)	RT_{NDT} BTP 5-3 B1.1(4) (°F)	T@50 ft-lb (L) (°F)	T@50 ft-lb (T) (°F)	RT_{NDT} NB-2331 (°F)	Error (°F)
1	5P-5933	Byron 1	40	0/102,105	0	-55.8	-33.9	40*	-40
2	411343	Catawba 1	-40	20/75,85	20	14.1	13.5	-40*	60
3	526840	McGuire 2	-4	0/92,96	0	-57	25.7	-4*	4
4	990400	N Anna 1	-12	40/66,78,84	40	20.4	82.4	22.4	17.6
5	990496	N Anna 2	-48	15/36,62	35	15.8	109.1	49.1	-14.1
6	3P2359	Oconee 2	20	0/48,55,60	0	-7.2	-6.1	20*	-20
7	4P1885	Oconee 2	-10	12/58,93,102	12	-6.2	8.3	-10*	22
8	522194	Oconee 3	20	30/52,62,72	30	22.5	15	20*	10
9	522314	Oconee 3	20	0/41,62,67	20	-23.5	30.5	20*	0
10	980919	Sequoyah 1	5	10/48,54,60	10	2.0	98.9	38.9	-28.9
11	288757	Sequoyah 2	-22	10/62,65,76	10	-24	42.4	-17.6	27.6
12	527536	Watts Bar 1	-22	15/71	15	-15.5	114.5	54.5	-39.5
13	527828	Watts Bar 1	14	0/30,71,116	20	-20.9	5.9	14*	6
14	49D867-1	Braidwood 1	-20	-10/80,97	-10	-41.2	-11.4	-20*	10
15	MK24-3	Braidwood 2	-30	30/68,71,77	30	2.4	8.1	-30*	60
16	49D330	Byron 2	-20	0/38,63	30	-9.1	-16.4	-20*	50
17	21918	P-Island 1	14	-10/56,64,70	-10	-6.5	4.1	14*	-24
18	22642	P-Island 2	-13	10/64,80	10	-6.9	34.5	-13*	23
								Average	6.87

Notes: (1) Nos. 1 to 13 are A508-2 forging; Nos. 14 to 18 are A508-3 forging
(2) RT_{NDT} with * means that T_{NDT} is controlling in determining the RT_{NDT} value

The AREVA SA508-2 data in the January 30, 2014, assertion will not be relied on for decision making either. This is because all data presented there (a total of approximately 35) has an average CVN energy at 10°F more than 45 ft-lb, suggesting that the data associated with average CVN energy at 10°F below 45 ft-lb have probably been excluded from the data analysis. Table 1 above shows that among the 18 datasets there are 4 (Nos. 2, 4, 8 and 15) having average CVN energy at 10°F less than 45 ft-lb such that use of Position B1.1(4) needs to go to higher test temperatures (20°F for No. 2, 40°F for No. 4, and 30°F for Nos. 8 and 15) to reach 45 ft-lb. Several CVN data performed between 10°F and 20°F are not included in this list of 4 because their test temperatures are close to 10°F. Based on this observation, it is highly unlikely that for a much larger AREVA dataset, there is not a single data having an average CVN energy at 10°F below 45 ft-lb. Table 1 also shows that Position B1.1(4) gives conservative estimates of RT_{NDT} for Data 2, 4, 8 and 15. This indicated that excluding data having average CVN energy at 10°F below 45 ft-lb from the dataset will artificially bias the trend towards the non-conservative side. This is why Table 1 shows an average error of 6.87 °F on the conservative side while the AREVA data shows an average error of 13.8 °F on the non-conservative side.

A plot of the data analysis based on the NRC TLR and the plot of the current analysis summarized in Table 1 are presented below for comparison. Figure 1 is a reproduction of Figure 10-4 from the NRC TLR. Figure 2 is a plot of the same data following Position B1.1(4) faithfully. Figure 2 indicates that many data points that were predicted by the NRC TLR approach in Figure 1 to be non-conservative (i.e., above the BTP 5-3 line) are now conservative based on the current approach. This supports the NRC staff's conclusion that the NRC TLR approach of using the Charpy curve over a range of test temperatures, instead of using the Charpy data at a single test temperature as required by Position B1.1(4), is overly conservative.

For plates, Figure 10-4 of the NRC TLR shows that the Position B1.1(4) line is bounding except for two data, which nevertheless are very close to the bounding line. Considering the above finding that the NRC TLR approach regarding Position B1.1(4) is overly conservative, the NRC staff has no doubt that Position B1.1(4) would bound all plate data if the same approach that was employed to produce Figure 2 results for forgings is employed for plates.

Based on the Table 1 results from the data analysis by NRC using the REAP database for forgings, the NRC staff determined that continued use of Position B1.1(4) to determine the mean initial RT_{NDT} values for plates and forgings is adequate. However, since Table 1 indicates that using a 2σ value of approximately 33.4 °C (60 °F) will bound all error (in the last column of Table 1) associated with using Position B1.1(4) for forgings, and since Figure 10-4 indicates that if Position B1.1(4) is followed faithfully, all plate data will be bounded, the staff concludes that for any applications using this position, an acceptable approach is to use a σ_I of 0 °F for plates and a σ_I of 30 °F for forgings. These σ_I values are combined with σ_Δ in accordance with RG 1.99, Revision 2 to calculate the margin in estimating the ART for irradiated materials.

Use of this σ_I value in the margin calculation for P-T limits and PTS evaluations will not be implemented in the proposed BTP 5-3 for 72 EFPYs because the risk-informed approaches in Enclosures 5 and 6 indicated that the resulting changes in risk considering this σ_I value do not meet the NRC criterion for implementing a new requirement.

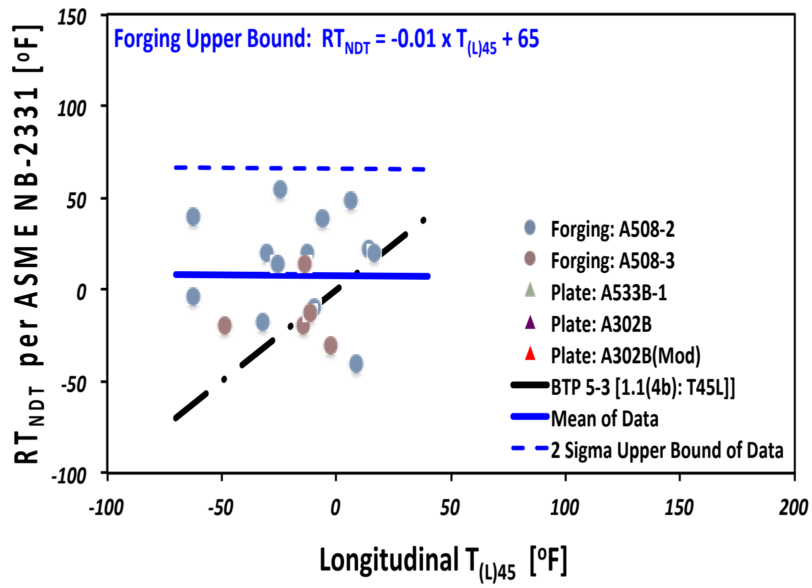

Figure 1 Data analysis results from Figure 10-4 of TLR-RES/DE/CIB-2014-011

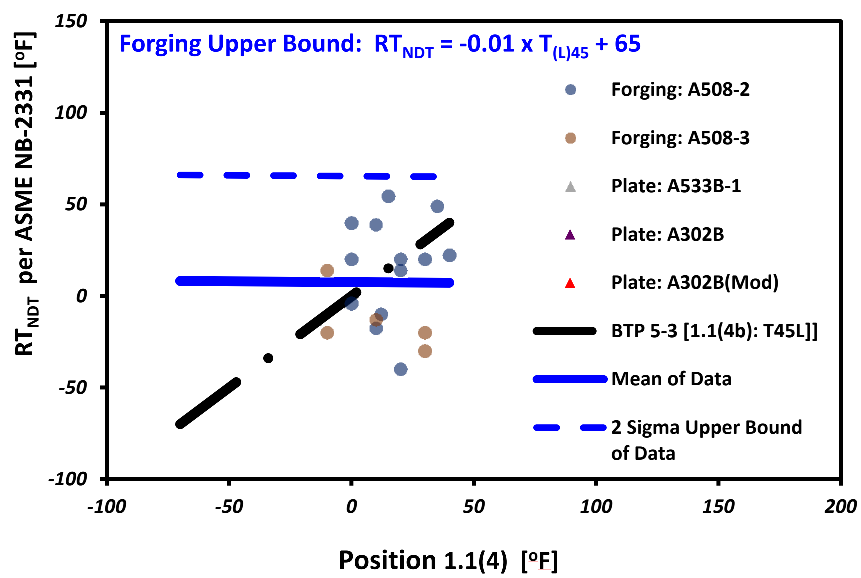

Figure 2 Current data analysis results from Table 1

3.1.3 Positions 1.2

This section presents industry's comparison of the transverse USE and the longitudinal USE for CVN data in the MRP-401 database. The NRC staff reviewed this information and that in Section 11 of the NRC TLR and concludes that the data analysis results from these two sources are consistent with the 1990 evaluation (Ref. 13), indicating that Position 1.2 is conservative for approximately 85% of the data (close to the mean - 1σ approach). Regarding use of generic data in support of compliance with 10 CFR Part 50, Appendix G and 10 CFR 50.61 (Ref. 14), the NRC did not always recommend using the mean - 2σ (the lower value is conservative) or mean + 2σ (the higher value is conservative) approach to derive generic values for material properties important to the RPV structural integrity evaluations. For instance, RG 1.99, Revision 2 permits use of the copper or nickel value for an RPV weld based on the mean of the measured values for welds that used the weld wire heat number that matches the subject RPV weld. On the other hand, the ASME K_{IC} (static fracture toughness for crack initiation) curve represents approximately a 95% lower tolerance bound curve. Therefore, the approved RPV structural integrity evaluation methodologies, which were used to demonstrate compliance of 10 CFR Part 50, Appendix G and 10 CFR 50.61, have applied different levels of conservatism to different generically derived parameters. Using this system of different levels of conservatism to different parameters in RPV structural integrity evaluations, combined with explicit safety factors used in the fracture mechanics analysis and assumptions such as postulation of a surface flaw with a depth of one quarter of the RPV wall thickness in the Appendix G methodology, is supported by 40 plus years of operating experience without challenge of its adequacy during operation. Thus, the NRC staff recommends no revision for Position 1.2.

3.2 Assessment of Section 5, "Reactor Pressure Vessel Pressure-Temperature Limits"

This section based on PFM evaluations for PWRs is not reviewed because (1) the NRC staff's review of the PWROG report (Ref. 6), as documented in Enclosure 3, concluded that the current licensing basis (CLB) P-T limits are technically appropriate in light of the BTP 5-3 non-conservatism, and (2) the NRC staff's risk-informed analyses documented in Enclosure 6 concluded that there is no need to consider BTP 5-3 non-conservatism in P-T limits evaluations for PWRs and BWRs for 72 EFPYs.

3.3 Section 6, "Reactor Pressure Vessel Pressurized Thermal Shock Evaluations"

Section 6 presents three examples of the PTS calculations based on the industry alternative that is discussed at the end of Section 3.1.1 of this assessment. Detailed information of the three examples are summarized in Table 6-1 to Table 6-3 of Section 6. Among them, the key parameters, i.e., the initial RT_{NDT} and σ_I, are from Table 4-2 of Section 4. Table 6-1 presents PTS evaluation results for a sample RPV plate with a high RT_{PTS} value based on the existing Position B1.1(3)b and the corresponding industry alternative. The NRC staff identified this plant to be the most limiting plant if its 50.61 PTS evaluation is revised to consider the non-conservatism in BTP 5-3. Table 6-2 presents PTS evaluation results for a sample RPV non-Rotterdam forging with a low RT_{PTS} value based on the industry alternative regarding Position B1.1(3)a (i.e., Case 2 of Table 4-2). Table 6-3 is of a different nature and is for two different RPV Rotterdam forging materials: one with a CLB initial RT_{NDT} value and one with an initial RT_{NDT} value determined by industry alternative Position B1.1(3)a. Generally speaking, the NRC staff finds this section clear and appropriate.

In addition to the above observations, the NRC staff concludes that this section has provided good examples on how RT_{PTS} can be updated considering the industry alternative for considering the non-conservatism in Position B1.1(3)a or Position B1.1(3)b. The NRC staff has established the validity of the data plots (Figure 4-1 to Figure 4-7) related to the 9 cases in Table 4-2. However, the accuracy of the regression equations and the standard errors have only been verified by the NRC staff for Cases 2 and 5 because Case 2 is used in an actual example discussed above, and Case 5 is used in the NRC staff's risk-informed evaluation regarding P-T limits in Enclosure 6.

The NRC staff performed similar plant-specific PTS evaluation for all 19 PWR plants which applied BTP 5-3 to their plates or forgings and found that no plant needs to consider the BTP 5-3 non-conservatism in its PTS evaluation for 60 years of operation. This plant-specific PTS evaluation is documented in Enclosure 4. The PTS evaluation for 72 EFPYs is based on change-in-risk assessments and is documented in Enclosure 5.

3.4 Section 7, "Impact on Other Minimum Temperature Requirements"

Section 7 presents the industry's evaluation of impact due to potential non-conservatism in BTP 5-3 on the minimum temperature requirements (i.e., the closure flange region requirements) of Table 1 of 10 CFR Part 50 Appendix G. The ASME Code, Section XI, Appendix G methodology was revised in 2002 to replace the crack arrest fracture toughness (K_{Ia}) curve with the plane strain fracture toughness (K_{IC}) curve to develop P-T limits for RPV beltline and extended beltline region with neutron fluence greater than $1x10^{17}$ n/cm^2 (E > 1 MeV). However, the minimum temperature requirements, which are based on the K_{Ia} curve, were not revised in 2002 and have not been revised since. Evaluation of the minimum temperature requirements based on the K_{IC} curve was finally provided in a draft NRC report, Draft TLR-RES/DE/CIB-2014-02 (Ref. 15). Reference 15 contains Table 4-2, listing the required minimum temperatures (14 minimum temperatures) for the closure flange region based on the K_{IC} curve and the K_{Ia} curve under normal operating conditions for a variety of RPV designs (7 designs). Table 4-3 provides the corresponding information (14 minimum temperatures) under leak test conditions.

Tables 7-1 and 7-2 of Section 7 provide similar information, which is identical to that based on an earlier version of Tables 4-2 and 4-3 that was published in the 2011 Pressure Vessel and Piping (PVP) Conference (Ref. 16). Although Tables 7-1 and 7-2 are based on the old NRC information, the impact is limited because only 4 out of 28 minimum temperatures are revised in Reference 15. Of the 4 affected minimum temperatures, only one (Table 7-1) is based on the K_{IC} curve. Tables 7-1 and 7-2 of Section 7 also reports available margin based on the K_{IC} curve to offset the potential non-conservatism caused by BTP 5-3. According to the updated NRC information in Reference 15, the available margin for Westinghouse 3 Loop in Table 7-1 should be revised from 120 °F to 112 °F. The available margins in Table 7-2 are not affected.

Consequently, the BTP 5-3 plants with Westinghouse 3 Loop design, i.e., North Anna 1 and 2, needs further examination. The staff verified that North Anna 1 and 2 replaced their RPV heads in 2003. The RPV heads were fabricated using the French Construction Code R-CCM 1993 Edition thru 1996 Addenda. Since this edition of the French Code requires transverse Charpy curve as the ASME Code after 1972, the non-conservatism in BTP 5-3 does not apply to the minimum temperature requirements based on the North Anna 1 and 2 RPV heads, even though their beltline regions are still affected.

Considering the available margins based on the K_{IC} curve in Section 7, as modified by the NRC staff discussed above, the NRC staff concludes that the segment of the current P-T limits based on the minimum temperature requirements of 10 CFR Part 50 Appendix G needs not be revised to consider the BTP 5-3 non-conservatism for PWR and BWR plants.

4.0 CONCLUSION

The NRC staff performed an assessment of the MRP-401 report, and, based on the technical assessment in Section 3, the NRC staff determined that the specific approaches in the report as discussed in the assessment are acceptable for considering the non-conservatism in BTP 5-3. However, use of these σ_I values in the margin calculation for P-T limits and PTS evaluations will not be implemented in the proposed BTP 5-3 for 72 EFPYs because the risk-informed approaches in Enclosures 5 and 6 indicated that the resulting changes in risk considering these σ_I values do not meet the NRC criterion for implementing a new requirement.

5.0 REFERENCES

1. Proprietary MRP-401 (BWRVIP-287), "Assessment of the Use of NUREG-0800 Branch Technical Position [BTP] 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels," September 2015 (ADAMS Accession No. ML15265A040).

2. NUREG-0800 Standard Review Plan Branch Technical Position 5-3, Revision 2, "Fracture Toughness Requirements," March 2007.

3. NRC's letter to EPRI, "Request for the Electric Power Research Institute Reports on Evaluation of Branch Technical Position 5-3 Procedures, Assessment of potential Impact on Reactor Pressure Vessel Integrity, and Evaluation of General Electric Procedure," June 1, 2015 (ADAMS Accession No. ML15134A292).

4. AREVA's letter to NRC, "Potential Non-Conservatism in NRC Branch Technical Position 5-3," January 30, 2014 (ADAMS Accession No. ML14038A265).

5. TLR-RES/DE/CIB-2014-011, "Assessment of Predictions of RTNDT and Upper Shelf Energy Made Using Branch Technical Position 5-3," December 9, 2016 (ADAMS Accession No. ML16341B108).

6. PWROG-15003-NP, Revision 0, "Material-Orientation Toughness Assessment (MOTA) for the Purposes of Mitigating Branch Technical Position (BTP) 5-3 Uncertainties," June 2015 (ADAMS Accession No. ML15268A086).

7. *Code of Federal Regulations*, Title 10, Part 50, Appendix G, "Fracture Toughness Requirements."

8. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, NB-2331, "Material for Vessels."

9. ASTM International Standard E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

10. ASTM International Standard E 208, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels."

11. NRR comments on TLR-RES/DE/CIB-2014-011, February 3, 2015 (ADAMS Accession No. ML15035A159).

12. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.

13. Memorandum from C.Z. Serpan to C.Y. Cheng, USNRC, "Ratio of Transverse to Longitudinal Orientation Charpy Upper Shelf Energy," June 25, 1990.

14. *Code of Federal Regulations*, Title 10, Part 50, Section 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

15. Draft TLR-RES/DE/CIB-2014-02, “Technical Basis for 10 CFR 50 Appendix G, “Fracture Toughness Requirements”: Minimum Temperature Requirements,” not published yet.

16. Eric M. Focht, “Evaluation of the Minimum Temperature Requirements for the RPV Closure Heal Flange Region in 10 CFR 50 Appendix G,” PVP2011-57208, Proceedings of the ASME 2011 Pressure Vessel and Piping Division Conference, Volume 1: Codes and Standards, Baltimore, Maryland, July 17-21, 2011.

ENCLOSURE 3

TECHNICAL ASSESSMENT BY THE DIVISION OF ENGINEERING ON PWROG-15003-NP, REVISION 0 "MATERIAL-ORIENTATION TOUGHNESS ASSESSMENT (MOTA) FOR THE PURPOSES OF MITIGATING BRANCH TECHNICAL POSITION (BTP) 5-3 UNCERTAINTIES"

Contributor: David G. Dijamco, NRC/NRR/DE

TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING ON PWROG-15003-NP, REVISION 0, "MATERIAL-ORIENTATION TOUGHNESS ASSESSMENT (MOTA) FOR THE PURPOSES OF MITIGATING BRANCH TECHNICAL POSITION (BTP) 5-3 UNCERTAINTIES"

1.0 INTRODUCTION

By letter dated September 23, 2015, the Pressurized Water Reactor Owners Group (PWROG) sent as an attachment, report PWROG-15003-NP, Revision 0, "Materials-Orientation Toughness Assessment (MOTA) for the Purposes of Mitigating Branch Technical Position (BTP) 5-3 Uncertainties" (Ref. 1) to the U.S. Nuclear Regulatory Commission (NRC). The purpose of the transmittal of PWROG-15003-NP, which was sent for information only, is to assist the NRC's on-going effort related to the BTP 5-3 reassessment. The letter is a response to the NRC's official request for the report dated May 26, 2015 (Ref. 2).

The NRC took the action on the BTP 5-3 reassessment after the industry's and NRC's independent evaluations confirmed AREVA's January 30, 2014, assertion (Ref. 3) that using Position B1.1(4) of BTP 5-3 may be non-conservative. Assessment of PWROG-15003-NP and consideration of its technical contents along with information from NRC's independent evaluations and other industry reports on this subject are essential to the BTP 5-3 reassessment. This enclosure documents the NRC staff's assessment of the technical contents of PWROG-15003-NP.

2.0 BACKGROUND

Appendix G, "Fracture Toughness Requirements," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (RCPB) of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation. The two most important material properties mentioned in Appendix G which affect fracture toughness of the RCPB ferritic materials, including the reactor pressure vessel (RPV), are nil-ductility transition reference temperature (RT_{NDT}) and upper-shelf energy (USE). They are determined in accordance with Section III, NB-2331 of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) (Ref. 4) and ASTM International Standard E 185-82 (Ref. 5). Determination of initial (i.e., unirradiated) RT_{NDT} also requires information on nil-ductility transition temperature (T_{NDT}), which is from the drop weight test per ASTM E 208 (Ref. 6).

Since plants using the ASME Code editions older than 1973 may not have complete test data to determine T_{NDT}, RT_{NDT}, or USE, BTP 5-3 was issued to provide alternates to determine them: Positions B1.1(1) and B1.1(2) are for T_{NDT}; Positions B1.1(3)a, B1.1(3)b, and B1.1(4) are for RT_{NDT}; and Position 1.2 is for USE.

For these older plants, BTP 5-3 has been used in pressure-temperature (P-T) limits, pressurized thermal shock evaluations, and flaw evaluations. However, AREVA informed NRC on January 30, 2014 (Ref. 3), that use of Position B1.1(4) in BTP 5-3 guidelines may be non-conservative. In response, the NRC has performed independent evaluations of the potential non-conservatism associated with the use of BTP 5-3. The NRC also engaged the industry in public meetings, held on June 4, 2014, February 19 and June 2, 2015, and January 19, 2016, to exchange information regarding resolution of this issue. The industry subsequently documented its evaluations of the uncertainties related to the use of BTP 5-3 in PWROG-15003-NP, Revision 0 and in a proprietary Electric Power Research Institute (EPRI) report, MRP-401/BWRVIP-287 (Ref. 7).

3.0 TECHNICAL ASSESSMENT

This assessment focuses on the key technical contents of PWROG-15003-NP. This assessment is not a safety evaluation. Therefore, there is no need for the NRC staff to document its review of Section 1, “Background,” and Section 6, “References.” There is also no need for the NRC staff to document its review of Section 5, “Summary and Conclusions,” because the NRC staff will rely on its own evaluations and conclusions summarized in Sections 3.1 to 3.3 to inform the BTP 5-3 reassessment.

3.1 Section 2, “Methodology and Approach

Section 2 includes a discussion of the BTP 5-3 methodology, the P-T limit curve development methodology, and the MOTA methodology. The NRC staff’s assessment focuses on the MOTA methodology because the BTP 5-3 methodology is a summary of the NRC’s BTP 5-3 positions and the P-T limit curve development methodology is based on NRC-approved topical report WCAP-14040-A, Revision 4 (Ref. 8).

The MOTA methodology described in PWROG-15003-NP is based on the premise that, with respect to a flaw, the fracture toughness of ferritic plates and forgings used for RPVs is dependent on the flaw orientation within the material. Specifically, the strong direction fracture toughness should be used for an axial flaw and the weak direction fracture toughness should be used for a circumferential flaw. This fracture toughness-to-flaw orientation correspondence is due to the fabrication process of ferritic plates and forgings used for RPVs. PWROG-15003-NP cites precedents in considering material orientation appropriate for the flaw evaluation. The NRC staff determined that two of them, NRC Regulatory Guide 1.161 (Ref. 9) and Welding Research Council (WRC) 175 (Ref. 10), are clear and convincing.

As discussed in Section 2 of this assessment, T_{NDT}, RT_{NDT}, and USE are important material parameters in RPV plates and forgings and are determined in accordance to ASME Code and ASTM requirements. For example, the ASME Code requires that CVN test specimens oriented in the weak direction be used, which eventually will determine the value of RT_{NDT} that is used in calculating fracture toughness, and ultimately, in the development of P-T limit curves. However, plants older than 1973 may not have the complete test data required by later editions of the ASME Code and ASTM, and therefore, BTP 5-3 is used as an alternative for determining T_{NDT}, RT_{NDT}, and USE. For example, BTP 5-3, Position B1.1(3)a, specifies that if CVN test specimens in the weak direction were not tested, the test results from CVN test specimens oriented in the strong direction, reduced to 65% of their value, can be used.

With regard to the ASME Code requirement of using CVN test specimens oriented in the weak direction, PWROG-15003-NP states that the ASME appeared to have taken the conservative approach of using weak properties for all materials and that WRC-175 did not recommend transverse (i.e., weak direction) testing for RPV shells.

The MOTA approach, therefore, takes into account the inherent conservativism in the P-T limit curve methodology, where axial flaws are evaluated using the weak fracture toughness value even though the flaw orientation allows the strong fracture toughness value to be used. As previously mentioned, P-T limit curves are based on the methodology of NRC-approved topical report WCAP-14040-A, Revision 4. These P-T limit curves use weak properties for both axial flaws and circumferential flaws, and PWROG-15003-NP states that the uncertainty in BTP 5-3 could only be an issue if it becomes so large that the P-T limit curve for the circumferential flaw becomes the limiting curve. The P-T limit curve for the axial flaw already has inherent margin because it is based on the weak fracture toughness value, when in actuality the strong fracture toughness value would apply for this orientation. PWROG-15003-NP determines the MOTA margin by developing a circumferential flaw P-T limit curve and iterating on the value of adjusted reference temperature (ART) until the curve intersects the axial flaw P-T limit curve which at low temperatures, is always dominant. ART is defined as the initial RT_{NDT} plus the change in RT_{NDT} plus margins due to uncertainties. PWROG-15003-NP then gives the following quantitative definition of MOTA margin:

> "The MOTA Margin is calculated by subtracting the Circumferential Flaw ART value from the original Axial Flaw ART value at the point of intersection."

This definition can be restated as the following, per the PWROG's presentation on the MOTA margin on February 19, 2015 (Ref. 11):

> "The MOTA Margin is defined as the ART difference between an Axial Flaw based P-T limit curve and a Circumferential Flaw based P-T limit curve."

The NRC staff finds this definition of MOTA margin acceptable. The MOTA margin presented in PWROG-15003-NP is applicable to PWR cylindrical vessel plates and forgings in the beltline and extended beltline base metal materials away from major discontinuities.

The NRC staff finds the bases for the MOTA approach acceptable because it is well-established that the fracture toughness of ferritic vessel plates and forgings, with respect to a flaw, is dependent on the flaw orientation, as evidenced from the cited precedents mentioned earlier. Furthermore, the NRC staff finds that the definition and method for calculation of MOTA acceptable because the method is based on the P-T limit methodology from NRC-approved topical report WCAP-14040-A, which is derived from the methodology of Appendix G to Section XI of the ASME Code.

3.2 <u>Section 3, "Discussion of Bounding Plant Geometries and Cases"</u>

PWROG-15003-NP determined MOTA margins for seven PWR plant types that include designs from the three PWR nuclear steam system suppliers: Westinghouse, Combustion Engineering, and Babcock & Wilcox. These seven representative designs include 2-loop, 3-loop, and 4-loop plants. One of the seven designs is a vessel that is thicker than the average vessels of the other designs. The cases used the current ART values for the actual P-T limit curves at each plant in most instances. However, when necessary, the actual plant ART values were increased to provide meaningful results to demonstrate the MOTA concept, i.e. if a given plant's

P-T limit curve was entirely dominated by the 10 CFR 50, Appendix G flange requirements. The NRC staff accepts increasing ART to clarify the MOTA concept because it removes the artificial notch on the P-T limit curves imposed by the flange requirements and thus produces smooth P-T limit curves that result strictly from the fracture mechanics analysis.

The NRC staff finds that including various PWR vessel geometries is reasonable for the purpose of evaluating the MOTA margin. Additionally, including a vessel thicker than the average PWR vessel thickness is a sound approach because the effect of larger thermal stresses due to the thicker section can be evaluated.

3.3 Section 4, "Review of Results"

PWROG-15003-NP presents P-T limit curves for steady-state, cooldown, and heatup conditions and determined the MOTA margin at the 1/4T and 3/4T locations, for each of the seven representative PWR geometries discussed in Section 3.2 of this assessment. PWROG-15003-NP reports lowest MOTA margins of 46°F for forgings and 40°F for plates. These MOTA margins are large enough to cover the increased margin due to a standard deviation for the initial RT_{NDT} (σ_I) of 20°F as discussed in Enclosure 2 for plates and non-Rotterdam forgings. However, for Rotterdam forgings, the MOTA margin of 46°F for forgings is not large enough to cover the increased margin due to a σ_I of 60°F as discussed in Enclosure 2. Consequently, the NRC staff examined current P-T limit information for the three plants having Rotterdam forgings based on BTP 5-3: North Anna 1 and 2 and Watts Bar 1. The NRC staff found that (1) each of the three plants has a very high ART value for the limiting forging for its current P-T limits, which is based on legitimate initial RT_{NDT} values per the ASME Code, Section III, (2) the current North Anna 1 and 2 P-T limits are based on a unique approach of considering a σ_I value of 30°F to account for use of BTP 5-3, and (3) the candidate Rotterdam forging in each plant having its initial RT_{NDT} value determined by BTP 5-3 will not become the new limiting material for the P-T limits even after considering the σ_I value of 60°F. Hence, the NRC staff concludes that although the MOTA margin of 46°F for forgings is not high enough to offset the BTP 5-3 non-conservatism for North Anna 1 and 2 and Watts Bar 1, the significant difference in the ART values between the limiting material and the candidate Rotterdam forging is large enough so that the Rotterdam forgings do not become the limiting material for the P-T limits for these plants.

PWROG-15003-NP also evaluated the effect of 10 CFR 50, Appendix G, Section IV.A.2.C requirements (the so-called "flange requirements") by artificially decreasing and increasing the original ART value for two of the representative PWR geometries. The MOTA margin is then determined for these cases. PWROG-15003-NP determined that the effect of the flange requirements on MOTA margin is non-existent for the 1/4T locations and negligible for the 3/4T flaw. This determination is reasonable since the moderator temperature in the P-T limit plot at which the circumferential and axial flaw curves intersect occurs at higher temperatures where the flange requirements become irrelevant.

The NRC staff performed calculations to confirm that the P-T limit curves for the 1/4T axial and circumferential flaw for Plant C are within the expected range of pressures and temperatures for PWRs. Hence, the MOTA margin results based on these P-T limit curves are reliable and can be used to inform the BTP 5-3 reassessment. Furthermore, the MOTA margin results reported in PWROG-15003-NP are relatively consistent across the seven representative PWR geometries.

4.0 CONCLUSION

The NRC staff performed a technical assessment of PWROG-15003-NP, which presents a methodology for determining the MOTA margin applicable to PWR cylindrical vessel plates and forgings in the beltline and extended beltline base metal materials away from major discontinuities. The NRC staff finds the bases for the MOTA approach reasonable because the approach considers precedents that take into account matching the flaw orientation with the corresponding material orientation. The NRC staff finds the definition and calculations of MOTA margin in PWROG-15003-NP correct, which uses P-T limit curves derived from the methodology of Appendix G to Section XI of the ASME Code. Consequently, the NRC staff concludes that the current licensing basis P-T limits are technically appropriate in light of the BTP 5-3 non-conservatism.

5.0 REFERENCES

1. Pressurized Water Reactor Owners Group (PWROG) report PWROG-15003-NP, Revision 0, “Materials-Orientation Toughness Assessment (MOTA) for the Purposes of Mitigating Branch Technical Position (BTP) 5-3 Uncertainties,” June 2015 (ADAMS Accession No. ML15268A086).

2. Letter from A. J. Mendiola (NRC) to W. A. Nowinowski (PWROG), “Request for the PWROG Report on Material Orientation Toughness Assessment for Mitigating Branch Technical Position 5-3 Uncertainties,” May 26, 2015 (ADAMS Accession No. ML15126A147).

3. Letter from P. Salas (AREVA) to the United States Nuclear Regulatory Commission, “Potential Non-Conservatism in NRC Branch Technical Position 5-3,” January 30, 2014 (ADAMS Accession No. ML14038A265).

4. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, NB-2331.

5. ASTM International Standard E 185-82, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels.”

6. ASTM International Standard E 208, “Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels.”

7. EPRI Technical Report, “Assessment of the Use of NUREG-0800 Branch Technical Position 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels,” MRP-401 and BWRVIP-287, September 2015 (PROPRIETARY).

8. Topical Report WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004 (ADAMS Accession No. ML050120209).

9. U. S. Nuclear Regulatory Commission Regulatory Guide 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less than 50 ft-lb,” June 1995 (ADAMS Accession No. ML003740038).

10. Welding Research Council Bulletin WRC-175, “PVRC Recommendations on Toughness Requirements for Ferritic Materials,” August 1972.

11. PWROG Presentation, “Material Orientation Toughness Assessment (MOTA) for the Purposes of Mitigating Branch Technical Position (BTP) 5-3 Uncertainties,” February 19, 2015 (ADAMS Accession No. ML15061A095)

ENCLOSURE 4

TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING ON THE PRESSURIZED THERMAL SHOCK AND PRESSURE-TEMPERATURE LIMITS FOR RELEVANT PWRS FOR 60 YEARS CONSIDERING NON-CONSERVATISM IN BTP 5-3

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TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING ON THE PRESSURIZED THERMAL SHOCK AND PRESSURE-TEMPERATURE LIMITS FOR RELEVANT PRESSURIZED-WATER REACTOR FOR 60 YEARS CONSIDERING NON-CONSERVATISM IN BTP 5-3

1.0 INTRODUCTION AND BACKGROUND

In response to AREVA's January 30, 2014, assertion that using Position B1.1(4) of NUREG-0800 Branch Technical Position (BTP) 5-3 may be non-conservative, the U.S. Nuclear Regulatory Commission (NRC) staff has reviewed all positions in BTP 5-3 to ensure that potential non-conservatisms in other positions of BTP 5-3 have also been evaluated. Similarly, industry performed an evaluation of this issue and produced two reports addressing the non-conservatism in applying BTP 5-3 positions: (1) MRP-401 (BWRVIP-287), "Assessment of the Use of NUREG-0800 Branch Technical Position 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels," September 2015 (Agency Document Access and Management System (ADAMS) Accession No. ML15265A040) and (2) Pressurized-Water Reactor Owners Group (PWROG)-15003-NP, Revision 0, "Material-Orientation Toughness Assessment (MOTA) for the Purposes of Mitigating Branch Technical Position (BTP) 5-3 Uncertainties," June 2015 (ADAMS Accession No. ML15268A086). This enclosure documents the NRC staff's plant-specific analyses for the relevant 19 pressurized-water reactors (PWRs), i.e., the plants which applied BTP 5-3 to some of their reactor pressure vessel (RPV) materials, to identify the plants that are affected by the non-conservatism in BTP 5-3.

These plants used the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) editions older than 1973, and therefore, do not have complete test data required by the newer ASME Code to determine nil-ductility transition temperature (T_{NDT}), nil-ductility transition reference temperature (RT_{NDT}), or upper-shelf energy (USE). As a result, BTP 5-3 was issued to provide alternative methods to determine these parameters: Positions B1.1(1) and B1.1(2) are for T_{NDT}; Positions B1.1(3)a, B1.1(3)b, and B1.1(4) are for RT_{NDT}; and Position 1.2 is for USE. The NRC staff examined these 19 PWRs and found that the majority of them applied Position B1.1(3). Although the specific BTP 5-3 positions that were applied to the remaining few plants cannot be confirmed, application of Position B1.1(3) was assumed in the NRC staff's plant-specific analyses for all 19 PWRs because it was demonstrated in Enclosure 2 that using Position B1.1(3) is more conservative than using other relevant positions. For simplicity, the RPV materials which applied BTP 5-3 positions to estimate their material properties will be referred to as the BTP 5-3 materials, or more specifically, as the BTP 5-3 plates or forgings, throughout this assessment.

2.0 REGULATORY ASSESSMENT

Appendix G, "Fracture Toughness Requirements," to Title 10 of *Code of Federal Regulations* (10 CFR) Part 50 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary (RCPB) of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation. Fracture toughness of the RCPB ferritic materials, including the RPV, depends on RT_{NDT} and USE. RT_{NDT} is typically determined using T_{NDT} and Charpy V-notch (CVN) curve from the impact tests in accordance with ASME Code, Section III, NB-2331, "Material for Vessels." RT_{NDT} is the key parameter in the fracture mechanic analyses associated with pressure-temperature (P-T) limits required by 10 CFR, Part 50, Appendix G; pressurized thermal shock (PTS) evaluations required by 10 CFR 50.61 or 50.61a; and flaw evaluations required by the ASME Code, Section XI. As such, BTP 5-3 has been used in these evaluations for the older plants with BTP 5-3 materials.

3.0 TECHNICAL ASSESSMENT

The NRC Staff's Screening Evaluation Based on Approach 1

To account for the non-conservatism in BTP 5-3, Enclosure 2 states that one acceptable approach for using Position B1.1(3) is to consider a standard deviation (σ_I) of 20 °F for all plates and non-Rotterdam forgings and a σ_I of 60 °F for Rotterdam forgings for their initial RT_{NDT} determination. As explained in Enclosure 2, this approach (Approach 1) is very conservative because (a) the "mean curve" is based on the more conservative BTP 5-3 line instead of the best fitted line, and (2) the σ_I is based on the line which bounds all data instead of the statistically determined value. Approach 1, though conservative, is convenient for screening out the affected plants. Use of Approach 1 is described below with results given in Table 1. Enclosure 2 further states that an acceptable alternative for Position B1.1(3) is listed in Table 4-2 of MRP-401, which uses both the mean and σ_I of the statistical analysis of a specific database of plates and forgings. Use of the acceptable alternative (Approach 2) is given in the first example under "Additional Plant-Specific Evaluations" further below.

When Approach 1 is applied, the specific σ_I value shall be combined with the standard deviation for the change of RT_{NDT} (ΔRT_{NDT}) due to neutron irradiation (σ_Δ) in accordance with Regulatory Guide 1.99, Revision 2 to calculate the margin in estimating the adjusted reference temperature (ART) for irradiated materials:

$$\text{Margin} = 2\,(\sigma_I^2 + \sigma_\Delta^2)^{1/2}$$

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

Currently, all 19 PWRs, except two, used a σ_I of 0 °F in the margin calculation for their ARTs. This gave a margin of 34 °F for plates and forgings for the case with no credible surveillance data (in this case σ_Δ is 17°F). However, if Approach 1 is used to account for the non-conservatism in BTP 5-3 (a σ_I of 20 °F for plates and non-Rotterdam forgings and a σ_I of 60 °F for Rotterdam forgings) with the σ_Δ of 17°F, the margin would become 52.5 °F for a plate or a non-Rotterdam forging and 124.7°F for a Rotterdam forging. Consequently, the ART would be increased by 18.5 °F for a plate or a non-Rotterdam forging and 90.7°F for a Rotterdam forging.

The objective of the NRC staff's screening evaluation was to identify among the 19 PWRs those which, after considering an extra margin of 18.5 °F for a plate or a non-Rotterdam forging and 90.7°F for a Rotterdam forging for the BTP 5-3 materials, would have an increased ART for the limiting material for P-T limits and an increased RT_{PTS} for PTS. For each such PWR identified, the limiting material is either the same BTP 5-3 material in the current licensing basis (CLB) or a BTP 5-3 material replacing the CLB limiting material (weld, plate, or forging). Table 1 listed the results of the NRC staff's screening evaluation. Two important outputs of the NRC staff's evaluation are the difference between the limiting ART and the highest ART of the BTP 5-3 materials for the 19 PWRs (Column 6 in Table 1) and the corresponding difference for RT_{PTS}

Table 1. The Difference (Safety Margin) Between the Limiting ART and the Highest ART of the Plates or Forgings Based on BTP 5-3 for 19 PWR RPVs

PWR	Plant	Reviewer	RPV Material Using BTP 5-3	P-T Limits		PTS		
				Impact to P-T Limits (CLB EFPY)	$(ART)_{limiting}$–$(ART)_{BTP5\text{-}3}$ (safety margin) (°F)	Impact to PTS (updated RT_{PTS} for 60 EFPY, °F)	Above PTS Limits	$(RT_{PTS})_{limiting}$–$(RT_{PTS})_{BTP5\text{-}3}$ (safety margin) (°F)
1	Calvert Cliffs 1	Widrevitz/ Sheng	Plate	no	49(3/4T)	No (254/AW)	no	84.5
2	Calvert Cliffs 2	Widrevitz/ Sheng	Plate	yes (33.1)	0[1]	Yes (198.2/P)	no	0
3	Cook 1	Widrevitz	Plate	no	65(3/4T)	No (254/AW)	no	124
4	ANO 2	Purtscher	Plate	yes (48)	0[1]	Yes (150.5/P)	no	0
5	Diablo Canyon 1	Poehler	Plate	no	52.7(3/4T)	No (239/AW)	no	94.9
6	Farley 1	Young	Plate	yes (54)	0[1]	Yes (220.5/P)	no	0
7	Fort Calhoun	Stevens	Plate	no	75 (3/4T)	No (268.1/AW)	no	124.3
8	Indian Point 2	Poehler	Plate	yes (48)	1[2] (1/4T)	Yes (269.7/P)	no[3]	0
9	Millstone 2	Purtscher	Plate	no	36(1/4T)	No (189/P)	no	38
10	North Anna 1	Purtscher	Rotterdam Forging	Yes (50.3)	85(3/4T)	Yes (200.7N[4])	no	46
11	North Anna 2	Purtscher	Rotterdam Forging	no	107(3/4T)	No (228/F)	no	124
12	Palisades	Young	Plate	no	60.7(1/4T) 38(3/4T)	No (approved 50.61a appl.)	no (but weld will exceed PTS limit)	68.9
13	Prairie Island 1	Stevens	Non-Rotterdam Forging	no	69(1/4T) 60(3/4T)	No (162/CW)	no	72.5
14	Salem 1	Young	Plate	no	51(1/4T) 29(3/4T)	No (267/AW)	no	64
15	Salem 2	Poehler	Plate	no	37(3/4T)	No (239/AW)	no	99
16	St. Lucie 1	Sydnor	Plate	no	39.8(3/4T)	No (234/AW)	no	86
17	TMI-1	Sydnor	Plate	yes (52)	0[2] (3/4T)	No (263.8/CW)	no	102
18	Waterford 3	Sydnor	Plate	yes (32)	0[1]	Yes (80.3/P)	no	0
19	Watts Bar 1	Sydnor	Rotterdam Forging	no	94.6(3/4T)	No (222.9/F)	no	125.1

Notes:

(1) The CLB limiting material and the material based on BTP 5-3 are the same material.
(2) The CLB limiting material and the material based on BTP 5-3 are different material.
(3) This RPV PTS evaluation used statistical mean and σ; all other RPV evaluations used only statistical σ.
(4) The BTP plate or forging becomes the new limiting material
(5) AW: axial weld; CW: circumferential weld; P: plate; and F: forging

(Column 9). These differences can be considered as "plant-specific safety margins against BTP 5-3 non-conservatism" (or simply "safety margins" hereafter in this assessment) for an RPV. Table 1 shows that 12 of the 19 PWRs have plant-specific safety margins greater than 18.5 °F for a plate or non-Rotterdam forging or 90.7°F for Rotterdam forging, and, therefore, will not be affected by the non-conservatism in BTP 5-3. For the remaining seven PWRs marked in yellow in Table 1, since the plant-specific safety margins (zero or nearly zero for six PWRs with BTP 5-3 plates) are insufficient to cover the non-conservatism in BTP 5-3, a generic safety margin, such as that provided by PWROG-15003-NP, is needed to ensure safe operation of the plants. The PWROG-15003-NP evaluation will be briefly discussed below, followed by three additional plant-specific evaluations worthy of discussion. For the NRC staff's assessment of this WCAP, please see Enclosure 3.

The PWROG-15003-NP Evaluation

The MOTA approach described in PWROG-15003-NP takes into account the inherent conservativism in the P-T limit curve methodology, where axial flaws are evaluated using the weak fracture toughness value even though the flaw orientation allows the strong fracture toughness value to be used. PWROG-15003-NP is limited to only P-T limits. The NRC staff's assessment of PWROG-15003-NP confirmed the existence of a MOTA margin of 40 °F for plates (the least among five PWR designs for the 1/4T and 3/4T locations) and 46 °F for forgings (the least among two PWR designs for the 1/4T and 3/4T locations) if the difference between fracture toughness along different orientations of the plate or forging is considered. It is obvious that these additional MOTA margins of PWROG-15003-NP are large enough to bound non-conservatism of 18.5 °F for the six PWRs mentioned above due to the BTP 5-3 consideration.

For the remaining PWR, PWR 10 in Table 1, the non-conservatism to be addressed is 90.7°F for the Rotterdam forging. But, as indicated in Table 4-1 of PWROG-15003-NP, the additional MOTA margin for the forging is only 46 °F, which is insufficient to bound the non-conservatism. Resolution of this concern for PWR 10 is discussed below in NRC staff's additional plant-specific evaluations.

Additional Plant-Specific Evaluations

PWR 8

The PTS evaluation is the only concern for this PWR because the generic safety margin provided by PWROG-15003-NP for P-T limits bounds the non-conservatism in BTP 5-3. The NRC staff performed PTS evaluations for other PWRs, applying Approach 1 for Position B1.1(3)a or Position B1.1(3)b, and determined that these PWRs meet the PTS criteria for 60 years of operation. For PWR 8, the NRC staff, however, applied Approach 2 for Position B1.1(3)b to achieve the same determination. The following are the key parameters for the PTS evaluation of PWR 8:

Limiting BTP plate = B2002-3
Initial RT_{NDT} = 17.2 °F*
48 EFPY surface fluence = 1.906E19 n/cm^2
Chemistry Factor = 176
Fluence Factor = 1.176
ΔRT_{NDT} = 207 °F
σ_i = 15.1°F*

σ_Δ = 17 °F
Margin = 45.5 °F*
RT_{PTS} = 17.2 °F + 207 °F + 45.5 °F = 269.7 °F*

*Based on Approach 2 for Position B1.1(3)b

The initial RT_{NDT} of 17.2 °F is obtained by the NRC staff by applying the best-fit equation of Approach 2 for Position B1.1(3)b to the estimated temperature of 81 °F corresponding to 50 ft-lb of Charpy energy. The estimated temperature was calculated by adding 60 °F to the existing CLB initial RT_{NDT} of 21 °F based on WCAP-16752-NP, "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation" (ADAMS Accession No. ML090760605). This initial RT_{NDT} via best-fitting is consistent with that in MRP-401 Table 6-1. Since the equation coefficients for Position B1.1(3)a in Table 4-2 of MRP-401 are less than Approach 2 for Position B1.1(3)b, Approach 2 for Position B1.1(3)b gives higher initial RT_{NDT} and is bounding.

In summary, the RT_{PTS} value for PWR 8 will be below the PTS screening criterion at the end of the period of extended operation.

PWR 10

Table 1 indicated that for PWR 10, the safety margin is 85 °F for P-T limits and 46 °F for PTS, insufficient to bound the non-conservatism of 90.7 °F for the Rotterdam forging. Therefore, this PWR is worthy of a further plant-specific evaluation. Based on the license amendment request (LAR) submitted on July 1, 2004 (ADAMS Accession No. ML041950277), the ART at 3/4T for the limiting RPV material is 196 °F for 50.3 EFPY, and the corresponding ART for the competing BTP 5-3 forging is 111 °F, providing a safety margin for P-T limits of 85 °F. The NRC staff examined material information in the LAR and in the NRC reactor vessel integrity database (RVID) and found that, unlike other PWRs which used a σ_I of 0 °F for the BTP 5-3 plate or forging, PWR 10 used a σ_I of 30 °F for its BTP 5-3 forging, producing a margin of 69 °F. As a result, the non-conservatism for PWR 10 is no longer 90.7 °F (124.7 °F – 34 °F), but 55.7 °F(124.7 °F – 69 °F). This non-conservatism is bounded by the safety margin of 85 °F for P-T limits. Therefore, the P-T limits for PWR 10 will not be affected.

The following is the key parameters for the PTS evaluation of PWR 10:

Limiting BTP forging = nozzle shell Rotterdam forging of Heat 990286/295213
Initial RT_{NDT} = 6 °F
50.3 EFPY surface fluence = 0.2E19 n/cm^2
Chemistry Factor = 176
Fluence Factor = 0.571
ΔRT_{NDT} = 70 °F
σ_i = 60°F
σ_Δ = 17 °F
Margin = 124.7 °F
RT_{PTS} = 6 °F + 70 °F + 124.7 °F = 200.7 °F

The surface fluence was calculated by the NRC staff based on the 1/4T fluence listed in the LAR. The above BTP forging will become the new limiting material for PTS because the current

CLB RT_{PTS} is 191 °F for the lower shell forging of Heat 990400/292332 per the March 26, 2009, measurement uncertainty recapture (MUR) submittal (ADAMS Accession No. ML090900055). In summary, the P-T limits for PWR 10 will not be affected. For PTS, although the nozzle shell Rotterdam forging of Heat 990286/295213 will become the new limiting material with RT_{PTS} of 200.7 °F at the end of the period of extended operation, this value is below the PTS screening criterion.

PWR 17

Table 1 indicated that for PWR 17, the safety margin is 0 °F for P-T limits and 102 °F for PTS, not consistent with the general understanding that these safety margins should not differ this much. Therefore, this PWR is worthy of a further plant-specific evaluation. Based on the LAR submitted on December 14, 2012 (ADAMS Accession No. ML12353A319), the ART at 3/4T for the limiting RPV material is 126.8 °F at 52 EFPY for the axial weld of Heat 8T1762 (WF-8). The other limiting material at 3/4T for the composite P-T limits is the circumferential weld of Heat 72105 (WF-70) with an ART of 178.5°F at 52 EFPY. Since the corresponding ART for the competing plate based on BTP 5-3 is 126.8 °F and the limiting P-T limit material for comparison with the BTP 5-3 plate is the axial weld, the safety margin for P-T limits is 0 °F.

For PTS, the January 8, 2008, license renewal application (ADAMS Accession No. ML080220252) reported that the RT_{PTS} values for the axial weld, circumferential weld, and the BTP 5-3 plate mentioned above are 197.6 °F, 263.8 °F, and 161.8 °F, giving a safety margin for PTS of 102 °F (263.8 °F - 161.8 °F). It should be noted that for PWR 17, the limiting PTS material is the circumferential weld, while the limiting P-T limit material is the axial weld. For P-T limits, the axial weld also became limiting for part of the P-T curves because the stress due to pressure is about two times the stress for a circumferential weld. In other words, the limiting material for PTS is determined by only material properties, but the limiting material for P-T limits is determined by both the material properties and the applied stresses. If the limiting material for PTS, i.e., the circumferential weld, was also the only limiting material for P-T limits, the safety margin for P-T limits would be 51.7 °F (178.5°F - 126.8 °F). A PWR with a safety margin for P-T limits of 51.7 °F and a safety margin for PTS of 102 °F would be very reasonable, considering the much smaller ΔRT_{NDT} for P-T limits due to neutron fluence attenuation in 3/4T of the RPV wall.

In summary, the seemingly inconsistency between the safety margins for PTS and P-T limits for PWR 17 is caused by the different limiting materials for PTS and P-T limits.

It should be noted that the deterministic approaches discussed in this section are based on the margin equation using appropriate σ_I values. However, use of these σ_I values in the margin calculation for P-T limits and PTS evaluations will not be implemented in the proposed BTP 5-3 for 72 EFPYs because the risk-informed approaches in Enclosures 5 and 6 indicated that the resulting changes in risk considering these σ_I values do not meet the NRC criterion for implementing a new requirement.

4.0 CONCLUSION

The NRC staff has reviewed the PTS and P-T limits for all 19 PWRs containing BTP 5-3 materials to assess the impact of non-conservatism in BTP 5-3 on these PWRs for 60 years of operation. This plant-specific evaluation is based on the docketed RPV structural integrity information for each plant, using Approach 1 or Approach 2 for Position B1.1(3). Considering (a) the screening results shown in Table 1, (b) further evaluations based on MOTA margins of PWROG-15003-NP, and (c) additional plant-specific evaluations, the NRC staff concludes that the combined plant-specific and generic safety margins in each PWR RPV bound the non-conservatism in BTP 5-3 and, therefore, there will be no impact on the PTS evaluation and P-T limits for 60 years of operation for the 19 plants with BTP 5-3 plates or forgings.

<u>ENCLOSURE 5</u>

TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING AND DIVISION OF RISK ASSESSMENT ON RISK ASSESSMENT OF PRESSURIZED THERMAL SHOCK EVENT FOR 72 EFPYS CONSIDERING NON-CONSERVATISM IN BTP 5-3

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TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING AND DIVISION OF RISK ASSESSMENT ON RISK ASSESSMENT OF PRESSURIZED THERMAL SHOCK EVENT FOR 72 EFPYS CONSIDERING NON-CONSERVATISM IN BTP 5-3

1.0 INTRODUCTION AND BACKGROUND

Nil-ductility transition reference temperature (RT_{NDT}) is a material property closely related to the temperature at which steel transitions from ductile to brittle behavior, and this temperature increases over time for steel which is exposed to neutron irradiation. The RT_{NDT} is a key input to two evaluations related to structural integrity of a reactor pressure vessel (RPV): the pressurized thermal shock (PTS) evaluation for pressurized water reactors (PWRs) required by Title 10, Part 50 of the *Code of Federal Regulations*, Section 50.61 (10 CFR 50.61) [1] or 50.61a [2] and the pressure-temperature (P-T) limits evaluation for both PWRs and boiling water reactors (BWRs) required by 10 CFR, Part 50, Appendix G [3].

In January 2014, AREVA notified the U.S. Nuclear Regulatory Commission (NRC) [4] of a potential non-conservatism in Position B1.1(4) (the fourth position) in Branch Technical Position (BTP) 5-3 [5] regarding determination of RT_{NDT} for RPV materials. The NRC staff confirmed this finding and performed an independent review of all positions in BTP 5-3. This issue potentially affects those plants which were granted construction permits prior to 1973 since these plants may not have performed all necessary materials tests to determine RT_{NDT} in accordance with later editions of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code), Section III, NB-2331 [6]. The NRC staff searched the RPV material information in the NRC's Reactor Vessel Integrity Database (RVID2) and found that 19 PWRs applied BTP 5-3 and a significant number of BWRs applied an owners group alternative which is similar to BTP 5-3, but slightly more conservative.

Due to a concern that an increased RT_{NDT} – after considering the non-conservatism in BTP 5-3 – may reduce the fracture toughness of the relevant RPV material and affect the P-T limit and PTS evaluations, the NRC staff performed probabilistic fracture mechanics (PFM) estimates for relevant PWRs and BWRs to evaluate the impact on P-T limits and PTS due to change in risk caused by this non-conservatism. Consistent with the guidance in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," the NRC staff performed a safety goal screening evaluation to determine whether further regulatory action would be appropriate.

This evaluation determined that requiring licensees to update their analyses to address the non-conservatism in BTP 5-3 would result in an increase in calculated risk due to more realistic input. This increase in calculated risk was below the screening guidelines contained in NUREG/BR-0058. Therefore, the NRC will not require current licensees to take additional actions. Instead, the NRC staff proposes to revise BTP 5-3 to reference the evaluation described herein to document the staff's decision-making rationale and ensure that licensees are aware of this issue and its resolution.

This enclosure focused on the PTS evaluations for relevant PWRs for 72 effective full power years (EFPYs) considering the non-conservatism in BTP 5-3 from the risk-informed perspective. These risk-informed studies leverage results of the NRC's PTS research program supporting 10 CFR 50.61a, including: NUREG-1806, "'Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61)" (the PTS Risk Study) [7 and 8] and NUREG-1874, ''Recommended Screening Limits for Pressurized Thermal Shock (PTS)'' [9]. The corresponding assessment regarding P-T limits for relevant PWRs and BWRs for 72 EFPYs are addressed separately in Enclosure 6.

2.0 REGULATORY ASSESSMENT

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. 10 CFR 50.61 requires that for each PWR, "the licensee shall have projected values of RT_{PTS} or RT_{MAX-X}, accepted by the NRC, for each reactor vessel beltline material." RT_{PTS} is RT_{NDT} evaluated for the end-of-license fluence for the RPV beltline materials using the formula in 10 CFR 50.61. RT_{MAX-X} is a new material property defined in 10 CFR 50.61a, which characterizes the RPV's resistance to fracture initiating from flaws found in material X (X can be axial weld, circumferential weld, plate, or forging). One of the key parameters for RT_{PTS} determination is initial RT_{NDT} of the RPV beltline material, which is determined in accordance with the ASME Code, Section III, NB-2331. For plants which do not have complete test data per the ASME Code, Section III, NB-2331 (usually older plants with construction permits granted prior to 1973), BTP 5-3 provides an alternative methodology to determine the initial RT_{NDT} values.

The staff used the guidance in NUREG/BR-0058 to determine whether to pursue a backfit which would require licensees to update their analyses and take additional actions if necessary. NUREG/BR-0058 has been applied primarily to define the risk reduction necessary to justify imposing a new requirement. Historically, a new requirement is unlikely to be implemented as a substantial safety improvement backfit if the associated risk reduction is below NUREG/BR-0058 thresholds. Similarly, it is unreasonable to revise BTP 5-3 in a way that would require licensees to consider the non-conservatism if the calculated risk increase is below NUREG/BR-0058 thresholds. The calculated risk increase is not a real risk increase, but an increase caused by a more realistic input which was not recognized in the past. Therefore, using the guidance in NUREG/BR-0058 is justified. NUREG/BR-0058, Figure 3.1, "Regulatory Analysis for Nuclear Power Plant Cost-Justified Substantial Safety Enhancements" and Figure 3.2, "Safety Goal Screening Criteria" state that, "with certain exceptions [office director determination based on strong engineering or qualitative justification]…regulatory initiatives involving new requirements to prevent core damage should result in a reduction of at least 1×10^{-5} in the estimated mean value CDF [core damage frequency]…in order to justify proceeding with further analysis."

The NRC staff also used guidance in NUREG-0800, "Standard Review Plan [(SRP)] for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" [10] to help structure its risk analysis. Although the SRP is typically used to review licensing actions requested by the licensee, it was considered to be useful in this case for evaluating the potential risk improvement associated with correcting the non-conservatism. SRP Chapter 19.2 and Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Section 2 [11] describe four elements contained in risk evaluations: (1) define the proposed changes, (2) conduct engineering evaluations, (3) develop implementation and monitoring strategies, and (4) document the evaluations and submit the request. The NRC staff opted to structure its risk analysis using these four elements.

Although not specifically identified in NUREG/BR-0058, the staff considered the five principles of risk-informed decision-making when performing its analysis:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

The objective of applying these principles was to provide a complete assessment which considered whether an alternative justification for the proposed new requirement, as described in NUREG/BR-0058, Revision 4, should be pursued.

3.0 TECHNICAL ASSESSMENT

3.1 Evaluation of PFM Analysis Results by Division of Engineering

The objective of the PTS evaluation is to ensure RPV integrity under postulated severe cool-down events when RPV pressure is still high. For each RPV beltline material, the conventional PTS evaluation of 10 CFR 50.61 calculates its RT_{PTS} at the end of its operation (typically 40 years of operation for the original license and 60 years for the renewed license) by summing the initial RT_{NDT}, the shift due to neutron embrittlement, and the margin term. Using this formula is a deterministic approach. To assess the impact on the PTS evaluation with and without consideration of the BTP 5-3 non-conservatism, the NRC staff relied on a probabilistic approach based on PFM analyses. Since a significant amount of PFM results are available in NUREG-1874 [9] which supported the 10 CFR 50.61a rulemaking, the NRC staff decided to use the key results of NUREG-1874 in the current BTP 5-3 PTS assessment to avoid unnecessary PFM analytical work. The NRC staff's approach is acceptable because the PFM results of NUREG-1874 are based on the most up-to-date PFM methodology endorsed by the NRC.

Specifically, the NRC staff's PFM analyses (the NRC staff's analyses or the BTP 5-3 study) used the estimated through-wall crack frequency (TWCF) as a measure of the risk of RPV failure. TWCF values are output from the PFM computer code: Fracture Analysis of Vessels - Oak Ridge (FAVOR) [12]. FAVOR is an NRC-sponsored fracture mechanics code developed to perform deterministic and probabilistic fracture mechanics analyses for PWRs and BWRs subjected to various PTS transients and heat-ups and cool-downs. By comparing the TWCF values of the first four limiting plants among the 19 PWRs considering the BTP 5-3 non-conservatism with the TWCF values without considering the BTP 5-3 non-conservatism, the change in TWCF values for these limiting plants can be established. The four limiting plants selected by the NRC staff in this BTP 5-3 study are: Indian Point 2 (IP-2), Palisades, Watts Bar 1 (WB-1), and North Anna 1 (NA-1). IP-2 was selected because the plate based on BTP 5-3 (the BTP 5-3 plate) in its RPV has the highest RT_{PTS} value. Palisades was selected because it is the first plant receiving approval for its 50.61a submittal WCAP-17628-NP, "Alternate Pressurized Thermal Shock (PTS) Rule Evaluation for Palisades" [13] with a BTP 5-3 plate having the second highest RT_{PTS} value. Further, the NRC staff can use the calculated results in the Palisades 50.61a submittal to validate the NRC staff's analyses. WB-1 was selected because the forging based on BTP 5-3 (the BTP 5-3 forging) in its RPV was fabricated by Rotterdam, having the largest discrepancy between actual and estimated RT_{NDT} values. NA-1 was selected because of the same reason as WB-1, but with the Rotterdam forging in the low fluence, traditional non-beltline region. The NRC staff believes that the change in TWCF values for these four limiting plants are representative for the group of PWRs most affected by this issue and is therefore consistent with the guidance in NUREG/BR-0058.

As mentioned before, the TWCF values in this assessment were not generated from direct FAVOR runs. Instead, they are inferred from the established correlations between TWCF values and new 50.61a parameters (RT_{MAX-AW} for axial welds, RT_{MAX-CW} for circumferential welds, RT_{MAX-PL} for plates and forgings, and RT_{MAX-FO} for forgings with underclad cracks). These correlations were based on calculated TWCF values (direct FAVOR runs) for three pilot plants: Beaver Valley 1 (BV-1), Palisades, and Oconee 1 (OC-1), as illustrated in Figure 3.12 in NUREG-1874 [9]. Each unit represents a different PWR vendor. Since our focus is the BTP 5-3 plates and forgings, the NRC staff only used the correlations related to plates and forgings to estimate the contribution to TWCF from the most limiting BTP 5-3 plate or forging in the RPVs of the four limiting plants. In the current assessment, the focus is RT_{MAX-PL} because the non-conservatism in BTP 5-3 only affects plates and forgings without underclad cracks.

3.1.1 The Spreadsheet Program to Estimate TWCF Values

To generate the TWCF values for the four limiting plants, the NRC staff developed a spreadsheet program entitled, "BTP 5-3 using 50.61a trend curve." This spreadsheet has programmed all equations in 10 CFR 50.61a to calculate ΔT_{30}, which is defined in 50.61a as, "the shift in the Charpy V-notch transition temperature at the 30 ft-lb energy level produced by irradiation." RT_{MAX-AW}, RT_{MAX-CW}, RT_{MAX-PL}, or RT_{MAX-FO} needed for the subsequent TWCF determination using the Figure 3.12 correlation is obtained by adding the calculated ΔT_{30} to the input initial RT_{NDT} of the subject beltline material (i.e., axial welds, circumferential welds, plates and forgings, or forgings with underclad cracks). In addition to copper content (Cu %), nickel content (Ni %), and fluence which are needed for the RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" [14] trend curve, the 50.61a trend curve also requires phosphorus (P) and manganese (Mn) contents, RPV cold leg temperature (T_C), and neutron flux information.

The second part of the spreadsheet has programmed the four correlations in Figure 3.12 of NUREG-1874 so that the TWCF for axial welds, circumferential welds, plates and forgings, or forgings with underclad cracks can be estimated from the respective correlation based on RT_{MAX-AW}, RT_{MAX-CW}, RT_{MAX-PL}, or RT_{MAX-FO} that was calculated from the first part of the spreadsheet. The correlation equations can be found in NUREG-1874 and take the following form:

$$TWCF_{95-xx} = \exp \{m \times \ln(RT_{MAX-xx} - RT_{TH-xx}) + b\}$$

With the correlation coefficients defined as:

Correlation coefficients	m	b	RT_{TH} (Rankine)
RT_{MAX-AW}	5.5198	-40.542	616
RT_{MAX-PL}	23.737	-162.36	300
RT_{MAX-CW}	9.1363	-65.066	616

The correlation equation for forgings with underclad cracks takes a different form:

$$TWCF_{95\text{-}FO} = 1.3x10^{-137} x\ 10^{0.185xRTMAX\text{-}FO}$$

BTP 5-3 Spreadsheet Program Validation

In addition to hand calculations verifying each term in 50.61a Equations 6 and 7 using RPV surveillance data from several plants, the BTP 5-3 spreadsheet program has been validated further in the following manner:

(1) The variables such as φt_e (related to neutron flux and time) and Cu_e (related to Cu) and the compound variable $f(Cu_e, P)$, which takes different values according to different ranges of φ, Cu, and Cu and P, have been validated by approximately 800 RPV surveillance data.

(2) The RT_{MAX-AW}, RT_{MAX-CW}, and RT_{MAX-PL} values calculated in the spreadsheet for Palisades have been verified against corresponding values in the 2014 Palisades 50.61a submittal [13] with complete agreement.

(3) The TWCF values from the spreadsheet correlation equations have been verified by a couple of applications based on WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval" [15]. This WCAP also used the correlation equations in NUREG-1874 to predict TWCF values.

(4) The spreadsheet program has been independently examined and validated by the NRC Office of Nuclear Regulatory Research.

Based on the verification and validation of the BTP 5-3 spreadsheet program, the NRC staff has confidence in using it to generate the change in TWCF values to support the current BTP 5-3 study.

3.1.2 The Assumptions Used in the BTP 5-3 Study

The following assumptions were used to prepare the input parameters for the four limiting plants in the BTP 5-3 study:

(1) All input parameters for Palisades are from its 2014 50.61a submittal in WCAP-17628 [13].

(2) All input parameters for IP-2, WB-1, and NA-1 are from their most recent license amendment request (LAR) submittals.

(3) Fluence values at 72 EFPYs are estimated from linear extrapolation from the fluence at the LAR EFPY.

(4) Phosphorus and manganese contents that are not in the LAR are based on the conservative estimates provided in 10 CFR 50.61a.

(5) RPV cold leg temperatures (T_C) that are not in the LAR are based on the NRC surveillance material database that were used to develop the 10 CFR 50.61a trend curve. Based on the information in this database, the NRC staff estimated the T_C to be 529 °F for IP-2, 559 °F for WB-1, and 552 °F for NA-1. Each of these T_Cs represents the average of the T_Cs for the RPV surveillance data of IP-2, WB-1, and NA-1.

(6) The standard deviation (σ_i) for the initial RT_{NDT} of a specific plate or forging that was used to account for the non-conservatism in BTP 5-3 is based on the bounding estimates of Enclosure 2: a σ_i of 20 °F for plates and non-Rotterdam forgings and 60 °F for Rotterdam forgings.

3.1.3 The Change in TWCF values for the Four Limiting Plants

Using the input data based on the assumptions described in Section 3.1.2, the BTP 5-3 spreadsheet program has generated the change in TWCF values in Table 1.

Table 1 The NRC Staff's Change in TWCF Values for Four Limiting Plants with Embrittlement at 72 EFPYs

Plant Name	Initial RT_{NDT} Adjustment (Δ)	TWCF w/o BTP 5-3 Non-conservatism	TWCF with BTP 5-3 Non-conservatism	Change in TWCF Values
IP-2 (plate)	18.5 °F (1x Δ)	3.16×10^{-9}	9.00×10^{-9}	5.84×10^{-9}
	37 °F (2x Δ)		2.46×10^{-8}	2.14×10^{-8}
Palisades (plate)	18.5 °F (1x Δ)	2.67×10^{-10}	8.51×10^{-10}	5.84×10^{-10}
	37 °F (2x Δ)		2.58×10^{-9}	2.31×10^{-9}
WB-1 (forging)	90.7 °F (1x Δ)	6.59×10^{-16}	2.87×10^{-12}	2.87×10^{-12}
	136 °F (1.5x Δ)		7.70×10^{-11}	7.70×10^{-11}
NA-1 (forging)	90.7 °F (1x Δ)	2.52×10^{-14}	3.00×10^{-12}	2.97×10^{-12}
	136 °F (1.5xΔ)		7.81×10^{-11}	7.81×10^{-11}

3.1.4 The Applicability of Table 1 Results to New Materials in Old Plants

The Table 1 results were developed for the BTP 5-3 plates or forgings for the 19 PWRs. However, Table 1 results also apply to new BTP 5-3 plates or forgings in old plants with construction permits prior to 1973, which become RPV beltline materials when their neutron fluence values start exceeding 1×10^{-17} n/cm^2. The following are the bases for this conclusion:

For new BTP 5-3 plates

The initial RT_{NDT}, Cu %, Ni %, and neutron fluence used in the spreadsheet PFM analysis for the bounding IP-2 RPV plate are 21 °F, 0.25 %, 0.60 %, and 2.86×10^{19} n/cm^2. Figure 4-5 of MRP-401, "Assessment of the Use of NUREG-0800 Branch Technical Position 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels" [16] indicated that an estimated Charpy V-Notch (CVN) 50 ft-lb temperature of 125 °F bounds all, except one, plate data. This one data point can be discounted because it differs from the rest by 60 °F. Therefore, the worst possible estimated value for the initial RT_{NDT} of a new BTP 5-3 plate is 65 °F (125 °F – 60 °F), which is greater than the value of 21 °F for the bounding plate. However, considering the high Cu %, Ni %, and neutron fluence values of the bounding plate, the assumed high initial RT_{NDT} of 65 °F for a new BTP 5-3 plate (44 °F higher) can be easily compensated for by the much smaller shift due to a neutron fluence far below 2.86×10^{19} n/cm^2. For example, the shift due to neutron irradiation per 10 CFR 50.61a for this bounding plate is 229 °F for this neutron fluence, but only 138 °F for an assumed lower fluence of 2.86×10^{18} n/cm^2 (91 °F lower).

<u>For new BTP 5-3 forgings</u>

The initial RT_{NDT}, Cu %, Ni %, and neutron fluence used in the spreadsheet PFM analysis for the bounding NA-1 RPV forging are 6 °F, 0.16 %, 0.74 %, and 2.86×10^{18} n/cm^2. Figure 4-7 of MRP-401 [16] indicated that an estimated CVN 50 ft-lb temperature of 65 °F bounds all Rotterdam forging data. Therefore, the worst possible estimated value for the initial RT_{NDT} of a new BTP 5-3 Rotterdam forging is 5 °F (65 °F – 60 °F), which is bounded by the value of 6 °F for the bounding forging.

3.2 The NRC Staff's Change-In-Risk Evaluation by Division of Risk Assessment with Inputs from Division of Engineering

3.2.1 <u>Define the Proposed Change</u>

The NRC staff performed an analysis to determine the potential calculated risk increase that would be realized if licensees were required to update their PTS analyses to account for the non-conservatism associated with using BTP 5-3 to determine the initial RT_{NDT} values for BTP 5-3 plates and forgings.

3.2.2 <u>Conduct Engineering Evaluations</u>

According to the guidelines in RG 1.174 and SRP Chapter 19.2, the second element associated with a risk-informed application is an analysis of the proposed change using a combination of traditional engineering analysis with supporting insights from a risk assessment.

The objective of the NRC staff's analyses was to determine whether the non-conservatism in using BTP 5-3 resulted in an acceptably small change in risk such that the guidelines in NUREG/BR-0058 indicated that no further regulatory action is warranted. The NRC staff's analyses involved estimating the potential increase in risk caused by the non-conservatism in BTP 5-3. The increase in risk was evaluated against NUREG/BR-0058 criteria to determine if the values met the specified regulatory guidelines. The other key principles in RG 1.174 were also addressed in the evaluation. The intent was to demonstrate that the NRC staff's analyses are bounding and can apply to all relevant PWRs generically.

The engineering evaluations were based on the NRC staff's PTS Risk Study that is the technical basis underlying 10 CFR 50.61a [2].

3.2.2.1 Engineering Evaluation

The NRC staff's analyses included a PFM analysis of the effect of adopting the proposed NRC staff position on the frequency of RPV failure due to postulated PTS transients. RPV failure is defined for the purposes of this BTP 5-3 study as through-wall cracking of the RPV wall. The likelihood of RPV failure was postulated to increase with increasing time of operation due to a decrease in RPV fracture resistance from irradiation. Credible, postulated PTS transients that could potentially lead to RPV failure were considered to occur at the worst time in the life of the plant (as defined by level of RPV embrittlement). The PFM methodology allowed for the consideration of distributions and uncertainties in flaw number and size, material properties, accident transients, stresses, and the non-conservatism in BTP 5-3. The NRC staff's analyses evaluated the impact of not considering non-conservatism in BTP 5-3 on the four limiting plants defined in Section 3.1 of this assessment. The analysis results were used to estimate the potential increase in calculated risk (not increase in actual risk) that could be realized if licensees were required to address the non-conservatism in BTP 5-3.

Consistency with the PTS Risk Study

Since the NRC staff's analyses are based on the calculated TWCFs from the PTS Risk Study underlying 10 CFR 50.61a (specifically, the plot of TWCF versus RT_{MAX-PL} in Figure 3.12 in Reference 9), the NRC staff's analyses leveraged key assumptions and information about uncertainties gained from the previous FAVOR PFM analyses supporting 10 CFR 50.61a, including flaw size and distribution, accident transients, frequency of transients, and stress resulting from PTS transients, cladding, and welding (residual stresses). Therefore, the NRC staff's analyses have adequately considered the engineering variables in determining the risk of RPV failure in this BTP 5-3 study. Applying Figure 3.12 to the current plant-specific change-in-risk analyses for the four limiting plants is justified because the RT_{MAX-PL} values for the three pilot plants in Figure 3.12 bound approximately all four limiting plants in the NRC staff's analyses.

The RPV material properties and neutron fluence information for each of the four leading plants used in the NRC staff's analyses are based on current-licensing-basis (CLB) data from the most recent LAR applications, with the CLB fluence extrapolated to 72 EFPYs. Further, in addition to Cu and Ni, material fracture toughness is also determined by Mn and P based on Table 4 of 10 CFR 50.61a, and is therefore consistent with the PTS Risk Study.

Non-conservatism in BTP 5-3

Enclosure 2 identified that the non-conservatism in BTP 5-3 comes from underestimating the standard deviation (σ_i) for the initial RT_{NDT} determination based on generic test data for plates and forgings. One way to address the non-conservatism is to use a σ_i of 20 °F for plates and non-Rotterdam forgings and 60 °F for Rotterdam forgings. These σ_i values may not be accepted universally. However, they are conservative and appropriate to be used in the NRC staff's change-in-risk analyses. As explained earlier, the NRC staff did not conduct full-scale FAVOR runs that uses these σ_i values as direct input. Instead, the NRC staff used the margin calculation in RG 1.99, Revision 2 [14] to estimate the adjustment to be added to the initial RT_{NDT} to account for the non-conservatism in BTP 5-3 for the spread sheet analysis. The margin is defined in RG 1.99, Revision 2 as $2\,(\sigma_I^2 + \sigma_\Delta^2)^{1/2}$, where σ_Δ is the standard deviation for ΔRT_{NDT} (or the Charpy curve shift) due to embrittlement. Enclosure 4 contains details of this approach, which indicated that this adjustment is 18.5 °F for plates and non-Rotterdam forgings and 90.7 °F for Rotterdam forgings. The NRC staff's analyses incorporated these adjustment in the initial RT_{NDT} and RT_{MAX-PL} values for the four limiting plants to estimate their TWCF values using the spread sheet program.

3.2.2.2 Probabilistic Risk Assessment

PTS events were viewed as providing the greatest challenge to PWR RPV structural integrity and, therefore, the probabilistic risk assessment (PRA) had to estimate the frequency and severity of PTS transients. PTS transients are not normally modeled in PRAs, and the NRC staff's analyses for the four limiting plants are based on the same PTS transients and frequencies from the NRC PTS Risk Study. As part of the NRC PTS Risk Study, PRA models were developed by the NRC staff for each of the three pilot plants (Palisades, BV-1, and OC-1) using plant-specific information. These PRA models included an event tree analysis that defined the sequences of events that are likely to produce a PTS challenge to RPV structural integrity for each of the pilot plants. Several operational sequences can thermally shock the vessel, including: a break of the main steam line, secondary depressurization through a relief valve, a loss of coolant accident (LOCA), or extended injection of high-pressure water. During these events, water level in the primary system is restored since it will have dropped due to contraction resulting from overcooling. The temperature differential between the nominally ambient temperature emergency coolant water and the operating temperature of a PWR produces significant thermal stresses in the thick section steel wall of the RPV. These stresses could be high enough to initiate a running cleavage crack, a crack that could propagate all the way through the vessel. The PRA models binned individual event tree sequences with like characteristics into representative PTS transients [17, 18, and 19].

The results of the PRA in the PTS Risk Study included descriptions of each PTS transient from which the thermal-hydraulic (TH) characteristics of each transient can be developed, and estimates of the frequency with which each transient was expected to occur. The final transient frequency estimates were distributions (histograms) which represented the combined frequency, including uncertainties, of all the event tree sequences incorporated into each bin.

The transient frequencies were input into the FAVPOST module, the final, post-processing module in the FAVOR Code. This module combined the conditional initiation and through-wall cracking probabilities through a matrix multiplication with the frequency histograms for each PTS transient provided by the PRA analyses.

3.2.2.2.1 Estimating the Risk Associated with the Proposed Position

The likelihood of RPV failure was postulated to increase after consideration of the non-conservatism in using BTP 5-3 in combination with a decrease in RPV toughness due to irradiation of the RPV plates or forgings to 72 EFPYs.

This BTP 5-3 study assumed that a through-wall crack will lead to core damage and that core damage will lead to a large early release. As a result, changes in large early release frequency (LERF) will be the most limiting. While NUREG/BR-0058 guidelines do not indicate a specific requirement regarding the decrease in LERF which should be pursued as a potential backfit, the NUREG/BR-0058 guidelines do indicated that a conditional probability of early containment failure or bypass (CPCFB) of 0.1 is consistent with Commission guidance on containment performance for evolutionary designs. As a result, the staff assessed the BTP 5-3 non-conservatism – which is not expected to have a direct impact on containment performance – against a TWCF of 1E-6 per year. TWCFs which were less than 1E-6 per year would not warrant additional regulatory action. Using this acceptance criteria ensured that both the CDF and containment performance guidelines were considered. The equation in FAVPOST was used to estimate risk for plant j;

$LERF_j = CDF_j = TWCF_j = \Sigma\ IE_{ji} * CPF_{ji}$

where,

IE_{ji} is the initiating event frequency (events per year) for each of the i representative PTS transients for plant j developed during the PTS Risk Study. The PTS Risk Study developed full distributions for the frequency of each PTS transient bin. IE_{ji} does not change with or without considering the proposed position regarding BTP 5-3.

CPF_{ji} is the conditional probability of RPV vessel failure (conservatively assumed to occur if a through-wall crack develops) given the TH characteristics of each of the i representative PTS transients for plant j. As described above, the RPV material properties and the distribution of flaw sizes are those expected to exist at the end of plant j's operating life. The initial RT_{NDT} is the parameter that changes when the non-conservatism in BTP 5-3 is considered and, therefore, CPF_{ji} changes when the proposed position is implemented.

The NRC staff determined that the PRA models of PTS transient frequency, the IE_{ji} and CPF_{ji} parameters, and the above equation appropriately capture the significant contributors to risk from RPV failure and, therefore, fulfill the RG 1.174 guidance that the analysis is capable of modeling the impact of the proposed change for the purposes of this evaluation. The NRC staff thus concluded that the staff's BTP 5-3 study provided a reasonable or bounding estimate of the increase in calculated risk which would be realized if the BTP 5-3 non-conservatism was corrected through the backfit process.

3.2.2.2.2 Evaluation of PRA Technical Adequacy

Based on the PTS Risk Study's detailed review of past studies and operating experience, extensive interactions between the PTS analysis team and the plant personnel at all units, and the opportunity for the same team to benefit from the multiple plant study insights while performing all the analyses, the NRC has confidence that the PTS transient frequency results from the PRA analyses in the PTS Risk Study are sufficiently well developed to be used in the NRC staff's change-in-risk estimates to meet the guidelines in NUREG/BR-0058. For more information, please see the detailed evaluation of this subject in the safety evaluation (SE) enclosed in WCAP-16168-NP-A, Revision 3 [15].

3.2.2.2.3 Generic Applicability and External Events

During the development of the PTS Risk Study, the NRC staff investigated the applicability of the results from the three pilot plants to the operating fleet of PWRs. This investigation examined plant design and operational characteristics of five additional plants as described in Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants" [20]. The overall approach was to compare potentially important design and operational features (as related to PTS) of the other PWRs to the same features of the pilot plants to determine the extent these features are similar or different.

In 72 FR 56275 [21], the NRC staff reported its conclusion that the TWCF results from the PTS Risk Study can be applied to the entire fleet of operating PWRs. This conclusion was based on an understanding of characteristics of the dominant transients that drive their risk significance.

The generic evaluation revealed no design, operational, training, or procedural factors that could credibly increase the severity of these transients or the frequency of their occurrence in the general PWR population above the severity/frequency characteristics of the three pilot plants that were modeled in detail.

The detailed plant-specific PRAs in the PTS Risk Study evaluated the contribution of internal initiating events to TWCF. Separately, potential contribution of external initiating events to PTS risk was also evaluated by the NRC [22]. Based on the results of the PTS Generalization Study, the NRC staff has concluded that the PTS transient characteristics (both frequency and TH characteristics) are generically applicable for all similar plants (i.e., plants from the same vendor) in the fleet. Based on the results of the external events analyses, the NRC staff has also concluded that the contribution of external events to the change in risk has been adequately evaluated and that the contribution to risk from external events is equal to or less than the contribution for internal events.

3.2.2.2.4 Comparison with NUREG/BR-0058 Guidelines

The results of the NRC staff's change-in-risk analyses were summarized in Table 1 in Section 3.1.3 where the bounding increases in risk were reported as 2.14E-8/year, 2.31E-9/year, 7.70E-11, and 7.81E-11/year for IP-2, Palisades, WB-1, and NA-1, respectively, considering the extremely conservative case that the adjustment for the BTP 5-3 non-conservatism is 200 % of the required (i.e., increased by 37 °F) for plates and non-Rotterdam forgings and 150 % of the required (i.e., increased by 136 °F) for Rotterdam forgings. These change-in-risk increases are well below the guideline for a plant-specific backfit as described in NUREG/BR-0058 and discussed in Section 3.2.2.2.1 of this document.

As explained in NUREG-1874 [9], the TWCF results documented only considered the internal events PTS sequence frequency, but the NRC's external events analysis [22] suggests strongly that the actual contribution of external initiating events is much smaller than the contribution from internal events. Table 1 shows that the largest increase in LERF was estimated as 2.14E-8/year for IP-2. Therefore, the NRC staff concluded that the greatest change in risk associated with considering non-conservatism in BTP 5-3 at any relevant PWR is less than 5E-8/year. The NRC staff finds that this increase is very small and consistent with the intent of the Commission's safety goals.

3.2.3 Implementation and Monitoring

The third element in the RG 1.174 approach is to develop an implementation and monitoring program to ensure that no adverse safety degradation occurs because of the proposed changes. Based on more than 40 years of operating experience for the U.S. fleet of plants which applied BTP 5-3 to their RPV plates and forgings, the NRC staff has determined that existing programs that monitor operating experience and the ASME Code, Section XI, in-service inspection (ISI) programs that detect degradation indications are sufficient to identify any future need to revisit this assessment.

3.2.4 Submit Proposed Change

The fourth and final element in RG 1.174 approach is the development and submittal of the proposed change to the NRC. Since the recommended path forward is not to change the technical contents of BTP 5-3, no submittals of any nature related to BTP 5-3 is required or expected from the licensees as a result of its non-conservatism.

3.2.5 Conformance with the Principles of Risk-Informed Regulation

In addition to the four element approach discussed above, RG 1.174 states that risk-informed plant changes are expected to meet a set of key principles. This section summarizes these principles and the NRC staff findings related to the conformance of the NRC staff's analyses with these principles.

Principle 1 states that the proposed change must meet the current regulations unless it is explicitly related to a requested exemption or rule change. The current guidance in BTP 5-3 provides alternative methods to determine the initial RT_{NDT} values for RPV plates and forgings that do not have complete test data per the ASME Code, Section III, NB-2331. The NRC staff's risk-informed analyses justify the current guidance and, therefore, satisfy Principle 1.

Principle 2 states that the proposed change shall be consistent with the defense-in-depth philosophy. The proposed change is consistent with the defense-in-depth philosophy because there is no change in RPV design and, having plant-specific or integrated RPV surveillance programs for RPV material embrittlement monitoring and the ASME Code, Section XI, ISI program for flaw detection as integral part of defense-in-depth for the RPV materials, change in the robustness of the RPV or other systems at the plant is minimal. Therefore, the NRC staff concludes that, in total, the proposed position of not revising the technical contents of BTP 5-3 continues to provide reasonable assurance that RPV integrity will be maintained consistent with the philosophy of defense-in-depth, and Principle 2 is met.

Principle 3 states that the proposed change shall maintain sufficient safety margins. Section 12 of the PTS Risk Study concluded that the calculations demonstrate that PTS events are associated with an extremely small risk of RPV failure, suggesting the existence of considerable safety margin. Since the NRC staff's analyses are based on the calculated TWCF values from the PTS Risk Study, the "existence of considerable safety margin" remains to be true. Therefore, the NRC staff concluded that the proposed change maintains sufficient safety margins, and Principle 3 is met.

Principle 4 states that when proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goals. The NRC staff concluded that the greatest increase in LERF associated with not revising BTP 5-3 is less than 5E-8/year, as indicated in the NRC staff's analyses. The NRC staff found that this increase is very small and consistent with the intent of the Commission's Safety Goals. Therefore, Principle 4 is met.

Principle 5 states that the impact of the proposed change should be monitored using performance measurement strategies. Since the proposed change is no change to the technical contents of BTP 5-3, no additional monitoring other than those already in the regulations such as the RPV surveillance program and the ASME Code, Section XI, ISI program is necessary. Therefore, Principle 5 is met.

3.3 NRC Staff Findings

10 CFR 50.61a provides alternative requirements for protection against PTS events. The PTS Risk Study underlying 10 CFR 50.61a concluded that the risk of through-wall cracking caused by PTS events is much lower than previously estimated. The rule provided new PTS screening criteria that were selected based on an evaluation that indicated that, after applying these new, relaxed criteria, the TWCF due to a PTS event at any PWR would be less than 1E-6/year.

Since the NRC staff's analyses are based on the calculated TWCFs from the PTS Risk Study, the NRC staff's analyses inherited all assumptions and uncertainties in the FAVOR PFM analyses supporting 10 CFR 50.61a, including flaw size and distribution, accident transients, frequency of transients, and stress resulting from PTS transients, cladding, and welding (residual stresses).

Based on the results of the PTS Generalization Study, the NRC staff has concluded that the PTS transient characteristics (both frequency and TH characteristics) are generically applicable for plants from the same reactor vendor. RPV embrittlement is, however, material, operating history, and age specific. To address this concern, the NRC staff (1) selected the plants having the greatest embrittlement in BTP 5-3 plates and forgings (2) considered the embrittlement corresponding to 72 EFPYs, and (3) added 100 % additional adjustment for plates and 50 % additional adjustment for forgings in the NRC staff's analyses to bound all PWRs with BTP 5-3 plates and forgings.

With all the conservatisms mentioned above in the NRC staff's analyses, the resulting change in risk caused by non-conservatism in BTP 5-3 still meets the guidelines stated in NUREG/BR-0058 and the five principles of risk-informed regulation indicated in RG 1.174. Regarding the quantitative risk impact criteria, the NRC staff included the effect due to external events by doubling the NRC staff's calculated change-in-risk values of Table 1 for comparison. Therefore, the proposed position regarding BTP 5-3 provide an acceptable level of quality and safety.

4.0 <u>CONDITIONS AND LIMITATIONS</u>

This BTP 5-3 study is based on the neutron fluence corresponding to 72 EFPYs of operation. For future operation beyond 72 EFPYs, the licensees for PWR RPVs with BTP 5-3 plates or forgings need to address the BTP 5-3 non-conservatism in their PTS evaluations before their RPVs enter operation beyond 72 EFPYs, considering any new information such as excessive operating transients and unexpected embrittlement behavior for highly aged plants based on operating experience.

5.0 <u>CONCLUSION</u>

The NRC staff has found that the change-in-risk associated with not pursuing a backfit in association with the non-conservatism in BTP 5-3 regarding PTS is consistent with the guidance provided by NUREG/BR-0058. This conclusion applies to the BTP 5-3 plates or forgings for the 19 PWRs and new BTP 5-3 plates or forgings in old plants with construction permits prior to 1973 (potentially more than 19), which become RPV beltline materials when their neutron fluence values start exceeding 1×10^{-17} n/cm^2. Since this BTP 5-3 study is based on the neutron fluence corresponding to 72 EFPYs of operation, the licensees for PWR RPVs with BTP 5-3 plates or forgings need to address the BTP 5-3 non-conservatism in their PTS evaluations before their RPVs enter operation beyond 72 EFPYs, considering any new information such as excessive operating transients and unexpected embrittlement behavior for highly aged plants based on operating experience. The NRC staff has proposed a revision of BTP 5-3 as shown in Enclosure 7 to reference the memorandum that contains the assessment described herein, along with assessments in five other enclosures, to document the closure of this issue and the rationale for not pursuing a plant-specific backfit at this time.

6.0 REFERENCES

1. Title 10, Part 50.61 of the Code of Federal Regulations, “Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”

2. Title 10, Part 50.61a, of the Code of Federal Regulations, “Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.”

3. Title 10 *Code of Federal Regulations*, Part 50, Appendix G “Fracture Toughness Requirements.”

4. AREVA letter dated January 30, 2014, Regarding Non-Conservatism in Using One of the Positions in Branch Technical Position 5-3, “Fracture Toughness Requirements” (ADAMS Accession No. ML14038A265).

5. NUREG-0800 Branch Technical Position (BTP) 5-3 (formerly BTP 5-2), “Fracture Toughness Requirements” (ADAMS Accession No. ML070850035).

6. American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code, Section XI, Rules for Construction of Nuclear Power Plant Components, NB-2331, “Material for Vessels.”

7. NUREG-1806, “Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report,” August 2007 (ADAMS Accession Nos. ML072830076 and ML072830081).

8. NUREG-1806, “Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Appendices,” August 2007 (ADAMS Accession No. ML07282069).

9. NUREG-1874, “Recommended Screening Limits for Pressurized Thermal Shock (PTS), 2007 (ADAMS Accession No. ML070860156).

10. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Section 19.2, “Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance,” June, 2007 (ADAMS Accession No. ML071700658).

11. Regulatory Guide 1.174, Revision 1, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” November 2002 (Adams Accession No. ML023240437).

12. Letter Report, Oak Ridge National Laboratories/TM-2007/0030, “Fracture Analysis of Vessels” (FAVOR Code, Version 6.1).

13. WCAP-17628-NP, Revision 1, “Alternate Pressurized Thermal Shock (PTS) Rule Evaluation for Palisades,” June 2014 (ADAMS Accession No. ML14211A525).

14. Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” May 1988.

15. WCAP-16168-NP-A, Revision 3, “Risk-Informed Extension of the Reactor Vessel In-Service Inspection Interval” (ADAMS Accession No ML082820046).

16. MRP-401 (BWRVIP-287), “Assessment of the Use of NUREG-0800 Branch Technical Position 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels,” September 2015 (ADAMS Accession No. ML15265A040)

17. Letter Report, “Beaver Valley Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA),” March 3, 2005 (ADAMS Accession No. ML042880454).

18. Letter Report, “Palisades Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA)”, March 3, 2005 (ADAMS Accession No. ML042880473).

19. Letter Report, “Oconee Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA),” March 3, 2005 (ADAMS Accession No. ML042880452).

20. Letter Report, “Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants,” December 14, 2004 (ADAMS Accession No. ML042880482).

21. Federal Register Notice, (72 FR 56275) "Alternative Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," October 3, 2007 (ADAMS Accession No. ML072780354).

22. Letter Report, “Estimate of External Events Contribution to Pressurized Thermal Shock (PTS) Risk,” October 1, 2004 (ADAMS Accession No. ML042880476).

ENCLOSURE 6

TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING AND DIVISION OF RISK ASSESSMENT ON RISK ASSESSMENT OF OPERATION UNDER PRESSURE-TEMPERATURE LIMITS FOR 72 EFPYS CONSIDERING NON-CONSERVATISM IN BTP 5-3

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TECHNICAL ASSESSMENT BY DIVISION OF ENGINEERING AND DIVISION OF RISK ASSESSMENT ON RISK ASSESSMENT OF OPERATION UNDER PRESSURE-TEMPERATURE LIMITS FOR 72 EFPYS CONSIDERING NON-CONSERVATISM IN BTP 5-3

1.0 INTRODUCTION AND BACKGROUND

10 CFR 50 Appendix G [1] requires use of the American Society of Mechanical Engineers Boiler & Pressure Vessel Code (ASME Code), Section XI, Non-mandatory Appendix G [2] methodology to establish minimum pressure-temperature (P-T) limits as a function of the reactor pressure vessel (RPV) temperature, neutron fluence, operating conditions, and material properties. This Appendix G methodology is a deterministic fracture mechanics analysis with fracture toughness based on nil-ductility transition reference temperature (RT_{NDT}). RT_{NDT} is a material property closely related to the temperature at which steel transitions from ductile to brittle behavior, and this temperature increases over time for steel which is exposed to neutron irradiation.

In January 2014, AREVA notified the U.S. Nuclear Regulatory Commission (NRC) [3] of a potential non-conservatism in Position B1.1(4) (the fourth position) in Branch Technical Position (BTP) 5-3 [4] regarding determination of RT_{NDT} for RPV materials. The NRC staff confirmed this finding and performed an independent review of all positions in BTP 5-3. This issue potentially affects those plants which were granted construction permits prior to 1973 since these plants may not have performed all necessary materials tests to determine RT_{NDT} in accordance with later editions of the ASME Code. The NRC staff searched the RPV material information in the NRC's Reactor Vessel Integrity Database (RVID2) and found that 19 pressurized water reactors (PWRs) applied BTP 5-3 and a significant number of boiling water reactors (BWRs) applied a General Electric (GE) alternative which is similar to BTP 5-3, but slightly more conservative. The RPV plate or forging which used BTP 5-3 or the GE alternative will be referred to as the BTP 5-3 plate or forging throughout this assessment.

Due to a concern that an increased RT_{NDT} – after considering the non-conservatism in BTP 5-3 – may reduce the fracture toughness of the relevant RPV material and affect the P-T limit and pressurized thermal shock (PTS) evaluations, the NRC staff performed probabilistic fracture mechanics (PFM) estimates for relevant PWRs and BWRs to evaluate the impact on P-T limits and PTS due to change in risk caused by this non-conservatism. Consistent with the guidance in NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," [5] the NRC staff performed a safety goal screening evaluation to determine whether further regulatory action would be appropriate. This enclosure focused on the P-T limit evaluations for relevant PWRs and BWRs for 72 effective full power years (EFPYs) considering the non-conservatism in BTP 5-3 from the risk-informed perspective.

This evaluation determined that requiring licensees to update their analyses to address the non-conservatism in BTP 5-3 would result in an increase in calculated risk due to more realistic input. This increase in calculated risk was below the screening guidelines contained in NUREG/BR-0058. Therefore, the NRC will not require current licensees to take additional actions. Instead, the NRC staff proposes to revise BTP 5-3 to reference the evaluation described herein to document the staff's decision-making rationale and ensure that licensees are aware of this issue and its resolution.

2.0 REGULATORY ASSESSMENT

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. As mentioned earlier, 10 CFR Part 50, Appendix G requires that P-T limits for system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences be as conservative as limits obtained using the ASME Code, Section XI, Appendix G methodology. One of the key parameters for P-T limits is initial RT_{NDT} of an RPV beltline material, which is determined in accordance with ASME Code, Section III, NB-2331[6]. For plants which do not have complete test data per NB-2331, BTP 5-3 provides an alternative methodology to determine the initial RT_{NDT} values.

The staff used the guidance in NUREG/BR-0058 to determine whether to pursue a backfit which would require licensees to update their analyses and take additional actions if necessary. NUREG/BR-0058 has been applied primarily to define the risk reduction necessary to justify imposing a new requirement. Historically, a new requirement is unlikely to be implemented as a substantial safety improvement backfit if the associated risk reduction is below NUREG/BR-0058 thresholds. Similarly, it is unreasonable to revise BTP 5-3 in a way that would require licensees to consider the non-conservatism if the calculated risk increase is below NUREG/BR-0058 thresholds. For an RPV which is operated under the current licensing basis (CLB) P-T limits, the calculated risk increase due to BTP 5-3 non-conservatism is not a real risk increase, but an increase caused by a more realistic input which was not recognized in the past. Therefore, using the guidance in NUREG/BR-0058 is justified. NUREG/BR-0058, Figure 3.1, "Regulatory Analysis for Nuclear Power Plant Cost-Justified Substantial Safety Enhancements" and Figure 3.2, "Safety Goal Screening Criteria" state that, "with certain exceptions [office director determination based on strong engineering or qualitative justification]…regulatory initiatives involving new requirements to prevent core damage should result in a reduction of at least 1×10^{-5} in the estimated mean value CDF [core damage frequency]…in order to justify proceeding with further analysis."

The NRC staff also used guidance in NUREG-0800, "Standard Review Plan [(SRP)] for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" [7] to help structure its risk analysis. Although the SRP is typically used to review licensing actions requested by the licensee, it was considered to be useful in this case for evaluating the potential risk increase associated with considering the non-conservatism. SRP Chapter 19.2 and Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Section 2 [8] describe four elements contained in risk evaluations: (1) define the proposed changes, (2) conduct engineering evaluations, (3) develop implementation and monitoring strategies, and (4) document the evaluations and submit the request. The NRC staff opted to structure its risk analysis using these four elements.

Although not specifically identified in NUREG/BR-0058, the staff considered the five principles of risk-informed decision-making when performing its analysis:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored using performance measurement strategies.

The objective of applying these principles was to provide a complete assessment which considered whether an alternative justification for the proposed new requirement, as described in NUREG/BR-0058, Revision 4, should be pursued.

3.0 TECHNICAL ASSESSMENT

The NRC staff's risk assessment relies on the calculated through-wall crack frequencies (TWCFs) of representative PWRs and BWRs considering the BTP 5-3 non-conservatism and small surface breaking flaws. TWCF, which will be mentioned frequently later, is the product of multiplying conditional probability of failure (CPF) and the frequency of the loading condition (or event frequency).

The CPFs were obtained by applying the Fracture Analysis of Vessels - Oak Ridge (FAVOR) computer code, Version 16.1 [9]. FAVOR is an NRC-sponsored fracture mechanics code developed to perform deterministic and probabilistic fracture mechanics analyses for PWRs and BWRs subjected to various transients, including heat-ups and cool-downs. FAVOR 16.1 was used to perform the PFM analyses under P-T limit transients for PWRs and BWRs supporting the current change-in-risk evaluation. In this enclosure henceforth, PFM analysis also means FAVOR analysis.

The event frequency for PWR P-T limits are based on industry estimate of plant pressure relief system actuations for PWRs, such as low temperature overpressure protection (LTOP). The PFM analysis results are discussed in Section 3.1, and the event frequencies are discussed in Section 3.2.

3.1 Evaluation of PFM Analysis Results by Division of Engineering

3.1.1 The Industry's PFM Analysis Results

Electric Power Research Institute (EPRI) assessed change in CPF due to applying the current and an EPRI-modified BTP 5-3 in defining PWR P-T limits and reported their results in MRP-401/BWRVIP-287 [10]. Similar results for BWRs were presented by EPRI and Sartrex in a presentation during the ASME Code Working Group on Operating Plant Criteria (WGOPC) meeting on November 3, 2015 [11]. Both are for 60 years of operation. The NRC staff reproduced these results in Table 1. It should be noted, however, that the industry's BWR results in Table 1 were obtained using FAVOR Code, Version 15.1-beta (FAVOR 15.1-beta), while the industry's PWR results in Table 1 were obtained using FAVOR 12.1. FAVOR 15.1-beta has corrected an error in FAVOR 12.1 that affects BWR RPVs. The PFM results from Reference 11 indicated that the CPF for a BWR RPV could be more than one order of magnitude lower if FAVOR 15.1-beta is used.

3.1.2 The NRC Staff's PFM Analysis Results

The NRC staff relies on the PFM analyses performed by the Division of Engineering staff of the Office of Nuclear Regulatory Research for decision making. Therefore, only very limited evaluation of the industry's PFM analyses and results is provided below when the NRC staff discusses the discrepancies between the industry's and the NRC staff's PFM analysis results. The industry's PFM analysis results are referenced primarily to ensure that the industry's and the NRC staff's conclusions based on their respective PFM analysis results are consistent.

To perform PFM analyses with and without consideration of the BTP 5-3 non-conservatism, the NRC staff relies on FAVOR Code to generate the essential CPF values. For PWRs, the NRC staff has considered hypothetical cooldowns along P-T limits and actual cooldowns; for BWRs, the NRC staff has considered hypothetical cooldowns along P-T limits, actual cooldowns, hypothetical leak tests along P-T limits, and actual leak tests. All of these P-T limit transients are for embrittlement levels corresponding to 72 EFPYs. These P-T limit transients to be used in the PFM analyses are bounding according to prior PFM analyses supporting a proposed Appendix G rule change. The NRC staff's PFM results are summarized in Table 1, along with the industry's results. For details of the PFM analyses, including bounding material identification (Beaver Valley 1, Watts Bar 1, and Oyster Creek) and selection of the Beaver Valley 1 and Palisades RPV geometries for PWRs and the Oyster Creek RPV geometry for BWRs, please see Reference 12.

Table 4 in Enclosure 2 of Reference 12 has three CPF results for each limiting plate or limiting forging for the Beaver Valley 1 RPV model and the Palisades RPV model. The first CPF (Case A) is for an RPV under the CLB P-T limits, without considering the BTP 5-3 non-conservatism in the PFM analysis. The second (Case B) is for the same RPV under the CLB P-T limits, considering the BTP 5-3 non-conservatism in the PFM analysis (using a standard deviation, σ_i, for the initial RT_{NDT} values as a FAVOR input). The third (Case C) is for the RPV under the P-T limits modified to account for the BTP 5-3 non-conservatism, and also considering the BTP 5-3 non-conservatism in the PFM analysis. The NRC staff believes that if the BTP 5-3 non-conservatism is accounted for in the modified P-T limits, then one should not treat this RPV different from any RPV not having BTP 5-3 plates or forgings which normally set $\sigma_i = 0$ in the PFM analysis. In other words, the NRC staff suspects that Case C may have double counted the BTP 5-3 non-conservatism: once in the input modified P-T limits and once in the input σ_i for the initial RT_{NDT} values in the PFM analysis. Consequently, the NRC staff relied on the CPF results from Cases A and B to generate the change in TWCFs to assess the impact to operating plant P-T limits due to non-conservatism in BTP 5-3. Nonetheless, the NRC staff also checked the change in TWCFs based on the CPF results from Cases A and C and confirmed that they will not change the evaluation conclusion based on Table 1 results.

Table 1 indicated that all loading scenarios generated TWCFs less than $1x10^{-6}$ per reactor year (/yr). TWCF less than $1x10^{-6}$ /yr is determined by the NRC to be acceptable for through-wall cracks caused by PTS in the 10 CFR 50.61a Rulemaking [14]. The calculated risk values for all scenarios of Table 1 also consider the existence of small, surface breaking flaws on the inner surface of the vessel that were slightly deeper than the stainless steel cladding (i.e., flaws approximately ¼-inch deep). Such flaws have been observed in the RPV head cladding of one commercial BWR (Quad Cities, see NUREG-1796 [15]), attributed to intergranular stress corrosion cracking and possibly hot cracking. Destructive analysis of a canceled RPV, as reported in NUREG/CR-6817 [16], did not reveal the presence of small surface breaking flaws. However, based on expert elicitation performed as a part of NRC research activities, small

Table 1. Conditional Probability of Failure and TWCF Estimated for Hypothetic Cooldowns and Leak Test Along the P-T Limits With and Without Considering Non-conservatism in BTP 5-3 (including postulated small surface breaking flaws)

Reactor Type	Loading Condition	Material Type	Consideration of BTP 5-3 Non-conservatism	Conditional Probability of Failure, CPF		Frequency of Loading Condition, F (events/yr)	TWCF = CPF x F (/yr)	
				Industry (60 yrs)	NRC (72 EFPYs)		Industry (60 yrs)	NRC (72 EFPYs)
PWR [a]	100°F/hour Cooldown	Plate	No	1.84×10^{-6}	3.6×10^{-6}	$6\text{x}10^{-6}$ [b]	$\approx 1\times10^{-11}$	$\approx 2\times10^{-11}$
			Yes	1.81×10^{-6}	3.8×10^{-6}	$6\text{x}10^{-6}$ [b]	$\approx 1\times10^{-11}$	$\approx 2\times10^{-11}$
		Non-Rotterdam Forging	No	1.41×10^{-8}	-	$6\text{x}10^{-6}$ [b]	$\approx 1\times10^{-13}$	-
			Yes	6.57×10^{-8}	-	$6\text{x}10^{-6}$ [b]	$\approx 4\times10^{-13}$	-
		Rotterdam Forging	No	7.29×10^{-9}	1.8×10^{-6}	$6\text{x}10^{-6}$ [b]	$\approx 4\times10^{-14}$	$\approx 1\times10^{-11}$
			Yes	4.04×10^{-9}	3.7×10^{-6}	$6\text{x}10^{-6}$ [b]	$\approx 2\times10^{-14}$	$\approx 2\times10^{-11}$
BWR [a]	100°F/hour Cooldown	Plate	No	4.02×10^{-7}	1.59×10^{-9}	$< 1\text{x}10^{-7}$ [c]	$< 4\text{x}10^{-14}$	$< 2\text{x}10^{-16}$
			Yes	3.03×10^{-7}	6.58×10^{-8}	$< 1\text{x}10^{-7}$ [c]	$< 3\text{x}10^{-14}$	$< 7\text{x}10^{-15}$
	40°F/hour Leak Test		No	2.90×10^{-8}	4.60×10^{-6}	$1\text{x}10^{-3}$ [d]	$3\text{x}10^{-11}$	$5\text{x}10^{-9}$
			Yes	4.27×10^{-8}	5.76×10^{-6}	$1\text{x}10^{-3}$ [d]	$4\text{x}10^{-11}$	$6\text{x}10^{-9}$

Notes:

a. In PWRs, leak tests and heatup are less severe than cooldown; therefore, they are not evaluated separately. Industry PWR and BWR results are from Ref. 10 and Ref. 11. Both are for 60 years. NRC PWR and BWR results are for 72 EFPYs.
b. See Section 3.2.2.1 of this assessment.
c. BWRs operate at saturated conditions, so the probability of reaching (non-saturated) P-T limits is very small.
d. Estimated as 0.5/526 using a Bayes posterior distribution for p, based on the Jeffrey's non-informative prior, which is beta(x + ½, n - x + ½). The mean of the distribution is [13]:

$$(x + \tfrac{1}{2}) / (n + 1)$$

where: x = the number of observed failures
= 0, as there are no known BWR leak test violations
n = the number of trials
= 525, calculated as the number of leak test events for 35 U.S. BWRs operating an average of 30 years each and assuming 24-month fuel cycles

This provides an upper bound estimate of the probability of plant operators violating plant procedures (which is required for BWRs to follow the P-T limit curve during a leak test event).

surface breaking flaws were conservatively assumed to exist in the staff's calculations that provided the basis for the RT_{NDT} limits in the Alternate PTS Rule [17].

It should be noted that there are significant discrepancies between the industry's and the NRC staff's CPF values in Table 1. This is not surprising because of the following:

(1) The industry's FAVOR models represent actual RPVs with BTP 5-3 plates or forgings, while the NRC staff's FAVOR models represent three hypothetical bounding plants based on the highest RT_{PTS} value for plates or forgings among all PWRs and the highest ART value for plates among all BWRs (BWRs do not have beltline forgings) in RVID2.

(2) The industry's neutron fluence is based on the CLB EFPY for actual RPVs, while the NRC staff's neutron fluence is based on the 32 EFPY RVID2 values extrapolated to 72 EFPYs.

(3) The industry's models reflect the material properties for actual RPV beltline materials, while the NRC's models assume that the entire RPV beltline is made of the bounding BTP 5-3 plate or forging and use the material properties of this BTP 5-3 plate or forging.

The list is not complete, and each item above indicated that the NRC staff's modeling assumptions are more conservative than the industry's. That's why, for most cases in Table 1, the NRC staff's CPF values are higher than the industry's values. For the two exceptions for BWRs, since the industry and the NRC staff's TWCF values (< $1x10^{-13}$) are far below NUREG/BR-0058 thresholds, pursuing the reasons has no practical meaning. Considering all above, the NRC staff will end discussions on the differences between the industry's and the NRC staff's FAVOR models here because the focus of this enclosure is to examine whether the industry's and the NRC staff's conclusions based on their CPF values are consistent, not to document all the differences between the industry's and the NRC staff's FAVOR modeling.

The NRC staff recognizes that plants do not operate along the P-T limit curves during normal operation. Therefore, the NRC staff performed additional PFM analyses with the more realistic transients described in Section 3.2.2. Results for the "actual P-T limit transient" are summarized in Table 2. Unlike the TWCF results for 60 years by EPRI and Sartrex (Table 1), these results are for embrittlement levels that are representative of 72 EFPYs. The NRC staff's CPF values in Tables 1 and 2 are based on the PFM analyses and results documented in Reference 12. In addition to the modeling conservatisms mentioned above, the NRC staff used a very conservative approach in Reference 12 to account for the BTP 5-3 non-conservatism, i.e., applying a σ_i of 20 °F for plates and non-Rotterdam forgings and 60 °F for Rotterdam forgings in the PFM analyses for the initial RT_{NDT} of a plate or forging. Reference 12 showed that for a PWR with Rotterdam forging, this conservative approach of using the σ_i associated with the BTP 5-3 line to account for the non-conservatism has produced TWCF values above $1x10^{-6}$ /yr for one of the four actual P-T limit transients. Nonetheless, after using a more realistic approach based on regression analysis using statistical mean and σ_i of the dataset to account for BTP 5-3 non-conservatism (see Reference 10 for this approach), Table 7 (the last two rows) of Enclosure 3 of Reference 12 showed that the TWCF value for this specific case for Rotterdam forging becomes less than $1x10^{-6}$ /yr. The TWCF values for the Rotterdam forging for PWRs in Table 2 reflected the updated values based on the regression analysis. Therefore, the final industry's and the NRC staff's PFM analyses results in Tables 1 and 2 showed that all PWR and BWR cases have TWCF values less than $1x10^{-6}$ /yr.

Furthermore, the contributions to the highest TWCF values in Tables 1 and 2 are predominantly attributable to small, surface-breaking flaws that penetrate the cladding layer into the ferritic base material, which is a conservative assumption since no such flaws have been reported in the beltline region of any U.S. plants.

Table 2. Conditional Probability of Failure and TWCF Estimated During Actual P-T Limit Transient for 72 EFPYS (including postulated small surface breaking flaws)

Reactor Type	Loading Condition	Material Type	Consideration of BTP 5-3 Non-conservatism	Conditional Probability of Failure, CPF	Frequency of Loading Condition, F (events/yr)	TWCF = CPF x F (/yr)
PWR	Actual Cooldown	Plate	No	4.9×10^{-7}	1.0 [(a)]	4.9×10^{-7}
			Yes	6.3×10^{-7}	1.0 [(a)]	6.3×10^{-7}
		Rotterdam Forging	No	8.8×10^{-9}	1.0 [(a)]	8.8×10^{-9}
			Yes	2.2×10^{-7}	1.0 [(a)]	2.2×10^{-7}
BWRs	100°F/hour Cooldown Following Saturation Curve	Plate	No	1.5×10^{-7}	1.0 [(a)]	1.5×10^{-7}
			Yes	2.6×10^{-7}	1.0 [(a)]	2.6×10^{-7}
	Plant Procedure Leak Test		No	$0.0\times10^{+00}$	1.0 [(a)]	$0.0\times10^{+00}$
			Yes	9.0×10^{-15}	1.0 [(a)]	9.0×10^{-15}

Notes: (a) Assumes one event every 12 months.

Except for the case of BWRs under plant procedure leak test, the TWCF values in Table 2 under actual P-T limit transients are more limiting than the TWCF values in Table 1 under operation following the P-T limits. Therefore, Table 2 and the last two rows of Table 1 will be used as the basis for the NRC staff to assess the change in TWCFs for operating plants due to non-conservatism in BTP 5-3. The change in TWCF for an RPV under the actual P-T limit transients for BTP 5-3 plates can be obtained from the TWCF results in the last column of Table 2. For instance, the change in TWCF for a PWR with BTP 5-3 plates under the actual cooldown is 1.4×10^{-7}, which is the TWCF in Row 2 of Table 2 (6.3×10^{-7}) minus the TWCF in Row 1 of the table (4.9×10^{-7}). Change-in-TWCF values for other cases can be generated similarly.

3.1.3 The Applicability of Table 1 and Table 2 Results to New Materials in Old Plants

The Tables 1 and 2 results were developed for the 19 PWRs and BWRs with the BTP 5-3 plates or forgings. Since the highest TWCF value in Table 1 is $6\text{x}10^{-9}$ /yr, the margin of more than two orders of magnitude is sufficient to bound any possible initial RT_{NDT} value of a new BTP 5-3 plate or forging. Hence, any conclusion based on Table 1 results applies to new BTP 5-3 plates or forgings in old plants with construction permits prior to 1973, which become RPV beltline materials when their neutron fluence values start exceeding $1\text{x}10^{-17}$ n/cm^2. The following is the

NRC staff's bases to conclude that the Table 2 results also apply to the new BTP 5-3 plates or forgings in old plants.

For new PWR BTP 5-3 plates

The initial RT_{NDT}, copper content (Cu %), nickel content (Ni %), and neutron fluence at RPV inner surface that were used in the PFM analysis for the bounding Beaver Valley 1 RPV plate are 27 °F, 0.2 %, 0.54 %, and $1x10^{20}$ n/cm^2. Figure 4-5 of MRP-401 [10] indicated that an estimated Charpy V-Notch (CVN) 50 ft-lb temperature of 125 °F bounds all, except one, plate data. This one data point can be discounted because it differs from the rest by 60 °F. Therefore, the worst possible estimated value for the initial RT_{NDT} of a new BTP 5-3 plate is 65 °F (125 °F – 60 °F), which is greater than the value of 27 °F for the bounding plate. However, considering the high Cu %, Ni %, and neutron fluence values of the bounding plate, the assumed high initial RT_{NDT} of 65 °F for a new BTP 5-3 plate (38 °F higher) can be easily compensated for by the much smaller shift due to a neutron fluence far below $1x10^{20}$ n/cm^2. For example, at one quarter of the RPV wall thickness, the shift due to neutron irradiation per 10 CFR 50.61 for the bounding plate is 204 °F for this neutron fluence, but only 123 °F for an assumed lower fluence of $1x10^{19}$ n/cm^2 (81 °F lower).

For new PWR BTP 5-3 forging

The initial RT_{NDT}, Cu %, Ni %, and neutron fluence at RPV inner surface that were used in the PFM analysis for the bounding Watts Bar 1 RPV forging are 47 °F, 0.17 %, 0.8 %, and $1x10^{20}$ n/cm^2. Figure 4-7 of MRP-401 [16] indicated that an estimated CVN 50 ft-lb temperature of 65 °F bounds all Rotterdam forging data. Therefore, the worst possible estimated value for the initial RT_{NDT} of a new BTP 5-3 Rotterdam forging is 5 °F (65 °F – 60 °F), which is bounded by the value of 47 °F for the bounding forging. Please note that the bounding forging with the initial RT_{NDT} of 47 °F is not a BTP 5-3 forging. Reference 12 selected the bounding plate and forging, considering all RPVs, not just the RPVs with BTP 5-3 plates or forgings.

For new BWR BTP 5-3 plates

The initial RT_{NDT}, Cu %, Ni %, and neutron fluence at RPV inner surface that were used in the PFM analysis for the bounding Oyster Creek RPV plate are 3 °F, 0.27 %, 0.53 %, and $1.4x10^{19}$ n/cm^2. As stated above, the worst possible estimated value for the initial RT_{NDT} of a new BTP 5-3 plate is 65 °F. Considering that the GE alternative is more conservative than the BTP 5-3, this value becomes 55 °F, which is greater than the value of 3 °F for the bounding plate. However, considering the high Cu %, Ni %, and neutron fluence values of the bounding plate, the assumed high initial RT_{NDT} of 55 °F for the new BTP 5-3 plate (52 °F higher) can be easily compensated for by the much smaller shift due to a neutron fluence far below $1.4x10^{19}$ n/cm^2. For example, at one quarter of the RPV wall thickness, the shift due to neutron irradiation per 10 CFR 50.61 for the bounding plate is 169 °F for this neutron fluence, but only 69 °F for an assumed lower fluence of $1.4x10^{18}$ n/cm^2 (100 °F lower).

3.2 Evaluation of Change-In-Risk Results by Division of Risk Assessment with Inputs from Division of Engineering

3.2.1 Define the Proposed Change

The NRC staff performed an analysis to determine the potential calculated risk increase that would be realized by requiring licensees to update their P-T limit analyses to account for the non-conservatism associated with using BTP 5-3 to determine the initial RT_{NDT} values for BTP 5-3 plates and forgings.

3.2.2 Conduct Engineering Evaluations

According to the guidelines in RG 1.174 and SRP Chapter 19.2, the second element associated with a risk-informed application is an analysis of the proposed change using a combination of traditional engineering analysis with supporting insights from a risk assessment.

The objective of the NRC staff's analyses was to determine whether the non-conservatism in using BTP 5-3 resulted in acceptably small changes in risk such that the guidelines in NUREG/BR-0058 indicated that no further regulatory action is warranted. The NRC staff's analyses involved estimating the potential increase in risk caused by the non-conservatism in BTP 5-3. The increase in risk was evaluated against NUREG/BR-0058 criteria to determine if the values met the specified regulatory guidelines. The other key principles in RG 1.174 were also addressed in the evaluation. The intent was to demonstrate that the NRC staff's analyses are bounding and can apply to all relevant reactors generically.

3.2.2.1 Engineering Evaluation

The NRC staff's PFM analyses assessed the effect of adopting the proposed NRC staff position on the frequency of RPV failure (i.e., CPF) under the CLB P-T limits at 72 EFPY. RPV failure is defined for the purposes of this BTP 5-3 assessment as through-wall cracking of the RPV wall. The likelihood of RPV failure was postulated to increase with increasing time of operation due to a decrease in RPV fracture resistance from irradiation. The PFM methodology allowed for the consideration of distributions and uncertainties in flaw type (embedded and surface-breaking), number, and size, material properties, stresses, and the non-conservatism in BTP 5-3. The NRC staff's PFM analyses evaluated the impact of not considering non-conservatism in BTP 5-3 on the three selected RPVs mentioned in Section 3.1.2 of this assessment. The analysis results were used to estimate the potential increase in calculated risk that could be realized if licensees were required to address the non-conservatism in BTP 5-3.

Consistency with the PFM methodology supporting 10 CFR 50.61a

Since the NRC staff's analyses are based on the same PFM methodology supporting 10 CFR 50.61a, the NRC staff's analyses leveraged key assumptions and information about uncertainties regarding flaw size and distribution, embrittlement estimation, and stress resulting from the transients and cladding. In addition, the NRC staff's analyses conservatively considered small surface-breaking flaws on the inner surface of the RPV. Therefore, the NRC staff's analyses have adequately considered the engineering variables in determining the risk of RPV failure in this BTP 5-3 assessment.

The RPV material properties and neutron fluence information for each of the three selected plants used in the NRC staff's analyses are based on the Palisades, Beaver Valley, Oyster Creek RPV geometries using the most embrittled RPV information in the RVID2 database, with their neutron fluence values extrapolated to 72 EFPYs. As such, using the selected RPV models are more conservative than using any RPV from the license-amendment-request applications.

Non-conservatism in BTP 5-3

Enclosure 2 identified that the non-conservatism in BTP 5-3 comes from underestimating the σ_i for the initial RT_{NDT} determination based on generic test data for plates and forgings. One way to address the non-conservatism is to use a σ_i of 20 °F for plates and non-Rotterdam forgings and 60 °F for Rotterdam forgings. These σ_i values may not be accepted universally. However, they are conservative and appropriate to be used in the NRC staff's PFM analyses. In the NRC staff's FAVOR runs, the BTP 5-3 conservatism was accounted for by setting the σ_i values to those values discussed above.

Event frequency of the P-T limit transient

As mentioned earlier, TWCF is a product of CPF and the event frequency. The following provides details of the NRC staff's determination of the event frequencies used in the current assessment. Typical P-T limits with a LTOP system curve are depicted in Figure 1. Currently, all PWRs are automatically protected by a relief valve system at low temperatures using the LTOP system. LTOP systems were developed over concerns for the possible extension of a pre-existing flaw into a through-wall crack in the RPV due to overpressurization when operating at non-steady-state (lower) RPV coolant temperatures.

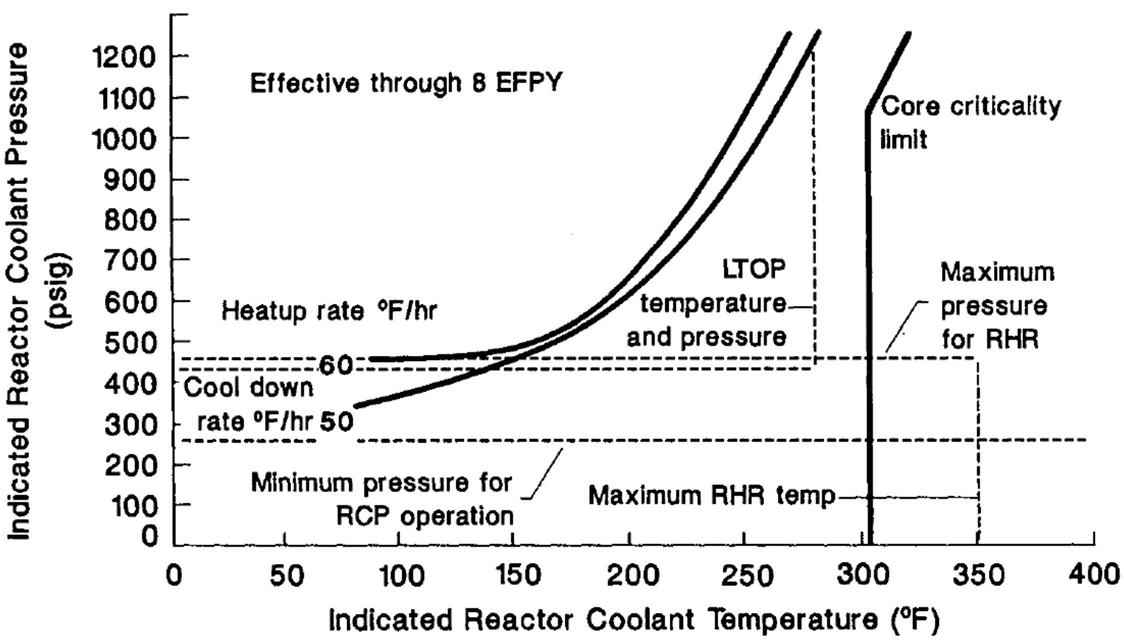

Figure 1. Typical Reactor Coolant System P-T Limits

The NRC has established processes and enforcement actions for assessing violations of plant Technical Specifications, including P-T limits. Actuations of plant pressure relief systems, such as LTOP, precede violation of P-T limits and do not necessarily indicate that a violation of the P-T limits has occurred. A recent industry estimate of plant pressure relief system actuations for PWRs, from 1984 through 2007, was 0.008 events/yr per the EPRI MRP-250 report [18]. This event frequency corresponded to those events where LTOP limits were exceeded, and the transient was limited to the LTOP pressure which are less than the P-T limits. As a result, the 0.008 events/yr frequency does not correspond to events that reach and exceed the P-T limits. Rather, events that reach and exceed the P-T limits correspond to those events where the LTOP system failed to actuate. Exceeding P-T limits at high temperatures is not relevant to this evaluation because RPV material is very tough at high temperatures, making it difficult to violate P-T curves.

From the probabilistic risk assessment (PRA) performed as a part of NRC PTS activities [19], the probability of the failure for one power operated relief valve (PORV) failing to open on demand at its set pressure is 9.29×10^{-3} (Note 1). There are two PORVs in a parallel circuit associated with PWR LTOP systems; therefore, failure of both PORVs to open is necessary for the reactor coolant system to experience an overpressure event during low temperature conditions, and the probability of such a failure is $(9.29 \times 10^{-3})^2 \approx 1 \times 10^{-4}$ per demand. Both PORVs must fail to open and the simultaneous failure probability must include common cause failures (a beta factor of 6.85×10^{-2} (Note 2)), which results in an additional probability of 6.36×10^{-4} ($9.29 \times 10^{-3} \times 6.85 \times 10^{-2}$) per demand. As a result, a conservative estimate of the frequency for reaching and exceeding the P-T limits at low temperatures is approximately $0.008 \times (6.36 \times 10^{-4} + 1 \times 10^{-4})$, or $5.89 \times 10^{-6} \approx 6 \times 10^{-6}$ events/yr.

The above paragraphs use plant configuration and industry event frequency to estimate the frequency of operating on the P-T curves. The reference industry event frequency is related to LTOP actuation which only occurs at the low temperature portions of the P-T curve. Event frequency at the high temperature portion of the P-T curve does not need to be considered because operation at the high temperature portion of the P-T curve has a much lower failure frequency than the lower temperature region.

For BWRs, the event frequency for BWR cooldown operations that follow the P-T curve is very low ($<1 \times 10^{-7}$) because BWRs follow steam saturated conditions, the conditions for which are well-below P-T limits. As is explained in footnote (c) of Table 1, the event frequency for BWR leak test that follows the P-T curve is 1×10^{-3}. Here, the NRC staff has completed presenting all event frequencies that were used in the NRC staff's TWCF calculations.

Actual PWR cooldown transients

The staff obtained more than fifty actual PWR cooldown transients that were provided to support the evaluation of Reference 18. From these cooldown events, four representative transients were selected for analysis. The staff also evaluated BWR cooldown events for operation that followed the water saturation curve at 100°F/hour, as well as a BWR leak test that followed typical procedural requirements. The event frequency for actual cooldown for PWRs or actual cooldown and leak test for BWRs is 1.0 which reflects the assumption of one event every year.

In addition to the above, information in Section 3.1 also supplements this engineering evaluation.

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1 See Table 8.1 of the PTS Probabilistic Risk Assessment (PRA) [19], where the probability of one PORV failing to open for Palisades is given as 9.29×10^{-3} (PRV-1042B POWER OPERATED RELIEF VALVE FAILS TO OPEN and PRV-1043B POWER OPERATED RELIEF VALVE FAILS TO OPEN).

2 See Table 37-1 of the Common Cause Failure Parameter Estimations [20], where the common cause failure fraction for 2 of 2 PORVs failing to open is given as 6.85×10^{-2}.

3.2.2.2 Probabilistic Risk Assessment

3.2.2.2.1 Estimating the Risk Associated with the Proposed Position

The likelihood of RPV failure was postulated to increase after consideration of the non-conservatism in using BTP 5-3 in combination with a decrease in RPV toughness due to irradiation of the RPV plates or forgings to 72 EFPYs.

This BTP 5-3 study assumed that a through-wall crack will lead to core damage and that core damage will lead to a large early release. As a result, changes in large early release frequency (LERF) will be the most limiting. While NUREG/BR-0058 guidelines do not indicate a specific requirement regarding the decrease in LERF which should be pursued as a potential backfit, the NUREG/BR-0058 guidelines do indicate that a conditional probability of early containment failure or bypass of 0.1 is consistent with Commission guidance on containment performance for evolutionary designs. As a result, the staff assessed the BTP 5-3 non-conservatism – which is not expected to have a direct impact on containment performance – against a TWCF of 1E-6 per year. TWCFs which were less than 1E-6 per year would not warrant additional regulatory action. Using this acceptance criteria ensured that both the CDF and containment performance guidelines were considered. To estimate risk for a plant under a P-T limit transient, the equation in FAVPOST (a post-processing module of the FAVOR Code) was used and simplified to:

LERF = CDF = TWCF = IE * CPF

where,

IE is the initiating event frequency (events per year) for the P-T limit transient. IE does not change with or without considering the proposed position regarding BTP 5-3.

CPF is the conditional probability of RPV vessel failure (conservatively assumed to occur if a through-wall crack develops) given the P-T limit transient. As described above, the RPV material properties and the distribution of flaw sizes are those expected to exist at 72 EFPYs. The σ_i associated with the initial RT_{NDT} is an input parameter that changes when the non-conservatism in BTP 5-3 is considered and, therefore, CPF changes when the proposed position is implemented.

Specifically, BTP 5-3, if revised, may cause changes to the CLB of certain PWR plants regarding P-T limits and PTS evaluations and certain BWR plants regarding P-T limits. This assessment focuses on the potential change in the risk resulting from applying a modified BTP 5-3 in establishing P-T limits. As discussed in Section 3.1.2, the risk associated with the proposed position is based on the limiting Table 1 and Table 2 TWCF results. In the current evaluation of potential change in the risk, TWCF is converted to a risk metric by conservatively equating the TWCF with LERF, i.e., assuming that a through-wall crack will consequently lead to a large early release, thereby ignoring the possibility that the crack might be relatively small and that mitigating systems might provide water to the vessel and might prevent or substantially delay any release from escaping containment.

The NRC staff determined that the IE and CPF parameters discussed in Section 3.2.2.1 and Section 3.1 respectively, and the above equation appropriately capture the significant contributors to risk from RPV failure and, therefore, fulfill the RG 1.174 guidance that the analysis is capable of modeling the impact of the proposed change for the purposes of this evaluation. The NRC staff thus concluded that the staff's BTP 5-3 study provided a reasonable or bounding estimate of the increase in calculated risk which would be realized if the BTP 5-3 non-conservatism was considered in the risk analysis.

3.2.2.2.2 Evaluation of PRA Technical Adequacy

Based on the review of past studies and operating experience, interactions between the analysis team and the plant personnel, and the opportunity for the team to benefit from the multiple plant study insights while performing the analyses, the NRC has confidence that the TWCF results are sufficiently well developed to be used in the NRC staff's change-in-risk estimates to meet the guidelines in NUREG/BR-0058.

3.2.2.2.3 Generic Applicability and External Events

The Risk Assessment of Operational Events Handbook, Volume 2 – External Events (RASP Handbook) represents best practices that the NRC staff use when performing risk assessments of operational events and licensee performance issues. The RASP Handbook can also be applied in the risk analyses for other regulatory applications and was reviewed in light of this evaluation to gage whether external events may be a significant contributing factor.

This external events that were considered for potential contributions to overall risk included: internal fires, internal flooding, seismic hazards, or other external events. The internal fire initiating event was judged not to be a dominant contributor of the overall risk from the non-conservatisms in BTP 5-3. The main reasons for this are because the non-conservatism is not related to degraded fire protection structures, systems, and components (e.g., fire suppression system, fire-rated barrier, smoke detection system) and its associated initiating event frequencies are not expected to be heavily influenced by postulated fire scenarios (e.g., risk-important cables running through the room of a redundant train). Similarly, the internal flooding, seismic hazard, and other external event initiating events were judged not to be dominant contributors of the overall risk from the non-conservatisms in BTP 5-3. However, the risk from each external event is expected to be slightly higher than the baseline due to the non-conservatism in BTP 5-3.

Because the initiating event frequency (Section 3.2.2.1) used for the purpose of this evaluation did not exclude external event initiators, there is no need to further consider the effects of the external events. Further, the most bounding increase in LERF was estimated to be 2.1E-7 per year, a factor of 4.76 below the NUREG/BR-0058 threshold value. Therefore, additional analysis to refine the contribution from external events is not warranted.

3.2.2.2.4 Comparison with NUREG/BR-0058 Guidelines

The results of the NRC staff's TWCF analyses were summarized in Tables 1 and 2 in Section 3.1. Using the conservative assumption that TWCF will lead directly to core damage and early release, the bounding increases in LERF were reported as 2.1E-7/year. These change-in-risk increases are well below the guideline for a plant-specific backfit as described in NUREG/BR-0058 and discussed in Section 3.2.2.2.1 of this document. The NRC staff finds that this increase is very small and consistent with the intent of the Commission's safety goals.

3.2.3 Implementation and Monitoring

The third element in the RG 1.174 approach is to develop an implementation and monitoring program to ensure that no adverse safety degradation occurs because of the proposed changes. Based on more than 40 years of operating experience for the U.S. fleet of plants which applied BTP 5-3 to their RPV plates and forgings, the NRC staff has determined that existing programs that monitor operating experience and the ASME Code, Section XI, in-service inspection (ISI) programs that detect degradation indications are sufficient to identify any future need to revisit this assessment.

3.2.4 Submit Proposed Change

The fourth and final element in RG 1.174 approach is the development and submittal of the proposed change to the NRC. Since the recommended path forward is not to change the technical contents of BTP 5-3, no submittals of any nature related to BTP 5-3 is required or expected from the licensees as a result of its non-conservatism.

3.2.5 Conformance with the Principles of Risk-Informed Regulation

The risk assessment called for evaluating the potential impact of any BTP 5-3 changes on the principles of risk-informed decision-making, to the extent they apply, or other factors that aid the decision process. The most likely proposed change if the risk associated with using the current P-T curves is not acceptable is to change BTP 5-3 (i.e., the current regulations). The following is NRC staff's risk assessment of BTP 5-3 non-conservatism along the five key principles of risk informed decision making:

Principle 1 states that the proposed change must meet the current regulations unless it is explicitly related to a requested exemption or rule change. The NRC staff's analyses using risk-informed approach demonstrated compliance with the current regulations even after considering the non-conservatism in BTP 5-3. Therefore, Principle 1 is satisfied.

Principle 2 states that the proposed change shall be consistent with the defense-in-depth philosophy. The proposed change is consistent with the defense-in-depth philosophy because there is no change in RPV design and, having plant-specific or integrated RPV surveillance programs for RPV material embrittlement monitoring and the ASME Code, Section XI, ISI program for flaw detection as integral part of defense-in-depth for the RPV materials, change in the robustness of the RPV or other systems at the plant is minimum. Therefore, the NRC staff concludes that, in total, the proposed position of not revising the technical contents of BTP 5-3 continues to provide reasonable assurance that RPV integrity will be maintained consistent with the philosophy of defense-in-depth, and Principle 2 is met.

Principle 3 states that the proposed change shall maintain sufficient safety margins. The staff assessment demonstrated that RPVs under the P-T limit transient are associated with an extremely small risk of RPV failure. Therefore, the NRC staff concluded that the proposed change maintains sufficient safety margins, and Principle 3 is met.

Principle 4 states that when proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goals. In this application, any proposed changes in BTP 5-3 result in lower fracture toughness of the relevant RPV plates and forgings and, thus, increase risk. However, it should be noted that this "increased risk" has existed in a RPV with BTP 5-3 plates or forgings since the plant's operation. It was recognized only recently when a more realistic representation of the initial RT_{NDT} values have been identified by the industry and the NRC. The risk evaluation conducting herein assumed that the TWCF is equivalent to the LERF. This conservative assumption negates any requirement for any additional PRA models. As described in Table 2, the estimated bounding TWCF is $6.3x10^{-7}$/yr for PWRs and $2.6x10^{-7}$/yr for BWRs based on very conservative assumptions (see Section 3.1). They are below the acceptable LERF value of 10^{-6}/yr that was developed in support of 10 CFR 50.61a. Further, the bounding change-in-risk is $2.1x10^{-7}$/yr for PWRs and $1.1x10^{-7}$/yr for BWRs, well below the guideline for a plant-specific backfit threshold for LERF of 1E-6/year in NUREG/BR-0058. Therefore, the NRC staff found that this increase is very small and consistent with the intent of the Commission's Safety Goals. Therefore, Principle 4 is met.

Principle 5 states that monitoring programs should be in place. P-T limits are continuously monitored in accordance with plant Technical Specifications for all leak test, cooldown, and heatup conditions. Therefore, Principle 5 is satisfied.

4.0 CONDITIONS AND LIMITATIONS

This BTP 5-3 study is based on the neutron fluence corresponding to 72 EFPYs of operation. For future operation beyond 72 EFPYs, the licensees for RPVs which applied BTP 5-3 or the GE alternative to their plates or forgings need to address the BTP 5-3 non-conservatism in their P-T limit evaluations before their RPVs enter operation beyond 72 EFPYs, considering any new information such as frequent unanticipated operation beyond P-T limits and unexpected embrittlement behavior for highly aged plants based on operating experience.

5.0 CONCLUSION

The NRC staff has found that the change-in-risk associated with not pursuing a backfit in association with the non-conservatism in BTP 5-3 regarding P-T limits is consistent with the guidance provided by NUREG/BR-0058. This conclusion applies to the BTP 5-3 plates or forgings for the 19 PWRs and BWRs and new BTP 5-3 plates or forgings in old plants with construction permits prior to 1973 which become RPV beltline materials when their neutron fluence values start exceeding $1x10^{-17}$ n/cm^2. Since this BTP 5-3 study is based on the neutron fluence corresponding to 72 EFPYs of operation, the licensees for RPVs which applied BTP 5-3 or the GE alternative to their plates or forgings need to address the BTP 5-3 non-conservatism in their P-T limit evaluations before their RPVs enter operation beyond 72 EFPYs, considering any new information such as frequent unanticipated operation beyond P-T limits and unexpected embrittlement behavior for highly aged plants based on operating experience. The staff proposes to revise BTP 5-3 as shown in Enclosure 7 to reference the memorandum that contains the assessment described herein, along with assessments in five other enclosures, to document the closure of this issue and the rationale for not pursuing a plant-specific backfit at this time.

6.0 REFERENCES

1. Title 10 *Code of Federal Regulations*, Part 50, Appendix G "Fracture Toughness Requirements."

2. American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Non-mandatory Appendix G, "Fracture Toughness Criteria for Protection Against Failure."

3. AREVA letter dated January 30, 2014, Regarding Non-Conservatism in Using One of the Positions in Branch Technical Position 5-3, "Fracture Toughness Requirements" (ADAMS Accession No. ML14038A265).

4. NUREG-0800 Branch Technical Position (BTP) 5-3 (formerly BTP 5-2), "Fracture Toughness Requirements" (ADAMS Accession No. ML070850035).

5. NUREG/BR-0058, Revision 4, "Regulatory Analysis Guideline of the U.S. Nuclear Regulatory Commission," U.S Nuclear Regulatory Commission, September 2004.

6. American Society of Mechanical Engineers (ASME) Boiler & Pressure Vessel Code, Section XI, Rules for Construction of Nuclear Power Plant Components, NB-2331, "Material for Vessels."

7. NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," June, 2007 (ADAMS Accession No. ML071700658).

8. Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002 (ADAMS Accession No. ML023240437).

9. Letter Report, Oak Ridge National Laboratories/TM-2007/0030, "Fracture Analysis of Vessels" (FAVOR Code, Version 16.1).

10. MRP-401/BWRVIP-287, "Assessment of the Use of NUREG-0800 Branch Technical Position 5-3 Estimation Methods for Initial Fracture Toughness Properties of Reactor Pressure Vessel Steels," September 2015.

11. EPRI and Sartrex, "Status of EPRI MRP/BWRVIP Appendix G Vessel Integrity Evaluations," the ASME Code Working Group on Operating Plant Criteria (WGOPC) meeting, Atlanta, GA, November 3, 2015.

12. NRC memorandum dated December 22, 2016, from Carol Nove to David Rudland, "Probabilistic Fracture Mechanics Evaluations Performed Using FAVOR to Assess the Risk Significance of the BTP-5.3 $RT_{NDT(U)}$ Estimation Errors" (ADAMS Accession No. ML16357A271).

13. NUREG/CR-6823, SAND2003-3348P, C. L. Atwood, et al., "Handbook of Parameter Estimation for Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, September 2003, p. 6-37.

14. NUREG-1806, Erickson Kirk, M. T., et al., "Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Limits in the PTS Rule (10 CFR 50.61): Summary Report," U.S Nuclear Regulatory Commission, August 2007.

15. NUREG-1796, ''Safety Evaluation Report Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2,'' U.S. Nuclear Regulatory Commission, October 31, 2004.

16. NUREG/CR-6817, Revision 1, F. A. Simonen, et al., "A Generalized Procedure for Generating Flaw Related Inputs for the FAVOR Code," U.S. Nuclear Regulatory Commission, October 2003.

17. Title 10, Part 50.61a, of the Code of Federal Regulations, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

18. EPRI Report No. 1016600, "Risk-Informed Method to Determine ASME Section XI Appendix G Limits for Ferritic Reactor Pressure Vessels: An Optional Approach Proposed for ASME Section XI Appendix G," MRP-250 and BWRVIP-215NP, Palo Alto, CA, 2009.

19. Letter Report, "Palisades Pressurized Thermal Shock (PTS) Probabilistic Risk Assessment (PRA)," ADAMS Accession No. ML042880473, March 3, 2005.

20. NUREG/CR-5497, NEEL/EXT-97-01328, "Common-Cause Failure Parameter Estimations," U.S. Nuclear Regulatory Commission, October.

ENCLOSURE 7

THE PROPOSED BTP 5-3 REVISION

Contributors: NRC/NRR/DE
NRC/NRR/DRA
NRC/NRR/DLR
NRC/RES/DE

U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

BRANCH TECHNICAL POSITION 5-3

FRACTURE TOUGHNESS REQUIREMENTS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of component integrity issues related to reactor vessels

Secondary - None

A. BACKGROUND

NRC requirements regarding fracture toughness, pressure-temperature limits, material surveillance, and pressurized thermal shock (PTS) (PWR only) are contained in Appendices A, G, and H to 10 CFR Part 50 and in 10 CFR 50.61; these requirements also refer to relevant sections of the ASME Code. The purpose of this branch technical position is to summarize these requirements and provide guidance, as necessary.

Since many of these requirements were not in force when some plants were designed and built, this position also provides guidance for applying these requirements to older plants. Also included is a description of acceptable procedures for making the conservative estimates and assumptions for older plants that may be used to show compliance with the new requirements.

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In 2014, a nuclear steam supply system vendor found that one of the estimates of the material property provided in B1.1 of this branch technical position is potentially non-conservative. In response, the industry has evaluated this position and some additional ones and the NRC has evaluated all positions under B1.1 and B1.2 for estimating specific material properties. This effort confirmed that varying degrees of non-conservatism exist in the four positions under B1.1. The NRC's assessment of these non-conservatisms and the impact of the non-conservatisms on operating plants regarding pressure-temperature limits and PTS was documented in a 2017 memorandum with enclosures assessing the industry's two summary reports on this issue and addressing all technical issues related to B1.1 and B1.2. Based on the change-in-risk analyses documented in this memorandum, the NRC determined that B1.1 and B1.2 can still be used in pressure-temperature limit and PTS evaluations up to 72 effective full power years.

B. BRANCH TECHNICAL POSITION

1. Preservice Fracture Toughness Test Requirements.

The fracture toughness of all ferritic materials used for pressure-retaining components of the reactor coolant pressure boundary shall be evaluated in accordance with the requirements of Appendix G, 10 CFR Part 50, as augmented by the criteria of Section III of the ASME Code. The fracture toughness test requirements for plants with construction permits prior to August 15, 1973 may not comply with the new Codes and Regulations in all respects. The fracture toughness of the materials for these plants should be assessed by using the available test data to estimate the fracture toughness in the same terms as the new requirements. This should be done because the operating limitations imposed on old plants should provide the same safety margins as are required for new plants.

1.1 Determination of RT_{NDT} for Vessel Materials

Temperature limitations are determined in relation to a characteristic temperature of the material, RT_{NDT}, that is established from the results of fracture toughness tests. Both drop weight nil-ductility transition temperature (NDTT) tests and Charpy V-notch tests should be run to determine the RT_{NDT}. The NDTT temperature, as determined by drop weight tests (ASTM E-208-1969) is the RT_{NDT} if, at 33 °C (60 °F) above the NDTT, at least 68 J (50 ft-lbs) of energy and 0.89 mm (35 mils) lateral expansion (LE) are obtained in Charpy V-notch tests on specimens oriented in the weak direction (transverse to the direction of maximum working).

In most cases, the fracture toughness testing performed on vessel material for older plants did not include all tests necessary to determine the RT_{NDT} in this manner. Acceptable estimation methods for the most common cases, based on correlations of data from a large number of heats of vessel material, are provided below for guidance in determining RT_{NDT} when measured values are not available.

(1) If dropweight tests were not performed, but full Charpy V-notch curves were obtained, the NDTT for SA-533 Grade B, Class 1 plate and weld

material may be assumed to be the temperature at which 41 J (30 ft-lbs) was obtained in Charpy V-notch tests, or -18 °C (0 °F), whichever was higher.

(2) If dropweight tests were not performed on SA-508, Class II forgings, the NDTT may be estimated as the lowest of the following temperatures:

(a) 33 °C (60 °F).

(b) The temperatures of the Charpy V-notch upper shelf.

(c) The temperature at which 136 J (100 ft-lbs) was obtained on Charpy V-notch tests if the upper-shelf energy values were above 136 J (100 ft-lbs).

(3) If transversely-oriented Charpy V-notch specimens were not tested, the temperature at which 68 J (50 ft-lbs) and 0.89 mm (35 mils) LE would have been obtained on transverse specimens may be estimated by one of the following criteria:

(a) Test results from longitudinally-oriented specimens reduced to 65% of their value to provide conservative estimates of values expected from transversely oriented specimens.

(b) Temperatures at which 68 J (50 ft-lbs) and 0.89 mm (35 mils) LE were obtained on longitudinally-oriented specimens increased 11 °C (20 °F) to provide a conservative estimate of the temperature that would have been necessary to obtain the same values on transversely-oriented specimens.

(4) If limited Charpy V-notch tests were performed at a single temperature to confirm that at least 41 J (30 ft-lbs) was obtained, that temperature may be used as an estimate of the RT_{NDT} provided that at least 61 J (45 ft-lbs) was obtained if the specimens were longitudinally oriented. If the minimum value obtained was less than 61 J (45 ft-lbs), the RT_{NDT} may be estimated as 11 °C (20 °F) above the test temperature.

1.2 Estimation of Charpy V-Notch Upper Shelf Energies

For the beltline region of reactor vessels, the upper shelf toughness must account for the effects of neutron radiation. Reactor vessel beltline materials must have Charpy upper shelf energy, in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 102 J (75 ft-lbs) initially and must maintain Charpy upper shelf energy throughout the life of the vessel of no less than 68 J (50 ft-lbs).

If Charpy upper shelf energy values were not obtained, conservative estimates should be made using results of tests on specimens from the first surveillance capsule removed.

If tests were only made on longitudinal specimens, the values should be reduced to 65% of the longitudinal values to estimate the transverse properties.

The predicted end-of-life Charpy upper shelf energy and adjusted reference temperature for the reactor vessel materials must meet the requirements of 10 CFR 50, Appendix G, paragraph IV.B. Reactor vessel materials that do not meet the specified end-of-life acceptance criteria are reviewed in accordance with paragraphs V.C and V.D of 10 CFR 50, Appendix G. NUREG-0744 provides an acceptable methodology for performance of fracture analysis for demonstrating adequate margins of safety for continued operation in accordance with 10 CFR Part 50, Appendix G, paragraph V.C.3.

1.3 Reporting Requirements

Fracture toughness information identified by the Code and by Appendix G, 10 CFR Part 50, should be reported in the FSAR to provide a basis for evaluating the adequacy of the operating limitations given in the Technical Specifications. In the case of older plants, the data may be estimated, using the procedures listed above, or other methods that can be shown to be conservative.

2. Operating Limitations for Fracture Toughness

2.1 Pressure-Temperature Operating Limitations

As required by Appendix G, 10 CFR Part 50, the following operating limitations shall be determined and included in the Technical Specifications. The basis for determination shall be reported, and is the responsibility of the applicant, but in no case shall the limitations provide less safety margin than those determined in accordance with Appendix G, 10 CFR Part 50, and Appendix G to Section III of the Code.

(1) Minimum temperatures for performing any hydrostatic test involving pressurization of the reactor vessel after installation in the system.

(2) Minimum temperatures for all leak and hydrostatic tests performed after the plant is in service.

(3) Maximum pressure-minimum temperature curves for operation, including startup, upset, and cooldown conditions.

(4) Maximum pressure-minimum temperature curves for core operation.

2.2 Recommended Bases for Operating Limitations

2.2.1 Leak and Hydrostatic Tests

(1) It is recommended that no tests at pressures higher than design pressure be conducted with fuel in the vessel.

(2) For system and component hydrostatic tests performed prior to loading

fuel in the reactor vessel, it is recommended that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60 °F.

(3) For system and component hydrostatic tests performed subsequent to loading fuel in the reactor vessel, the minimum test temperature should be determined as discussed in Section III of SRP 5.3.2.

2.2.2 Heatup and Cooldown Limit Curves

Heatup and cooldown pressure-temperature limit curves may be determined using single pr/t stress calculations, using the method given in Appendix G of the Code. The effect of thermal gradients may be conservatively approximated by the procedures in Appendix G of the Code or from the report, Tabulation of thermally-Induced Stress Intensity Factors (K_{IT}) and Crack Tip Temperatures for Generating P-T Curves per ASME Section XI-Appendix G, ORNL/NRC/LTR-03/03.

Calculations need only be performed for the beltline region, if the RT_{NDT} of the beltline is demonstrated to be adequately higher than the RT_{NDT} for all higher stressed regions.

Alternatively, more rigorous analytical procedures may be used, provided that the intent of the Code is met, and adequate technical justification for all assumptions and bases is provided.

2.2.3 Core Operation Limits

To provide added margins during actual core operation, Appendix G, 10 CFR Part 50 requires a minimum temperature during core operation, and a 22 °C (40 °F) margin in temperature over the pressure-temperature limits as determined for heatup and cooldown in 2.2.2 above. The minimum temperature, regardless of pressure, is the temperature calculated for the inservice hydrostatic test according to 2.2.1 above.

2.2.4 Upset Conditions

The pressure-temperature limits described in 2.2.2 and 2.2.3 above are applicable to upset conditions. Normal operating procedures should permit variations from intended operation, including all upset conditions, without exceeding the limit curves.

2.2.5 Emergency and Faulted Conditions

It is recognized that the severity of a transient resulting from an emergency or faulted condition is not directly related to operating conditions, and resulting temperature-stress relationships in the reactor coolant boundary components are primarily system dependent, and therefore not under direct control of the operator.

For these reasons, operating limits for emergency and faulted conditions are not a requirement of the Technical Specifications.

The SAR should present descriptions of the continued integrity of all vital components of the RCPB during postulated faulted conditions. It is recommended that such descriptions be made in as realistic a manner as possible, avoiding grossly over conservative assumptions and procedures.

2.3 Reporting Requirements

The Technical Specifications should include the operating and test limits discussed above, and the basis for their determination. The Technical Specifications should also include information on the intended operating procedures, and justify that adequate margins between the expected conditions and the limit conditions will be provided to protect against unexpected or upset conditions.

3. Inservice Surveillance of Fracture Toughness

The reactor vessel may be exposed to significant neutron radiation during the service life. This will affect both the tensile and toughness properties. A material surveillance program in conformance with Appendix H, 10 CFR Part 50, must be carried out.

3.1 Surveillance Program Requirements

The minimum requirements for the surveillance program are covered by Appendix H, 10 CFR Part 50. The selection of material to be included in the surveillance program should be in accordance with ASTM E-185-82, unless the intent of the program is better realized by using more rigorous criteria. For example, the approach of estimating the actual RT_{NDT} and upper shelf toughness of each plate, forging, or weld in the beltline as a function of service life, and choosing as the surveillance materials those that are expected to be most limiting, may be preferable in some cases. This would include consideration of the initial RT_{NDT}, the upper shelf toughness, the expected radiation sensitivity of the material (based on copper and nickel content, for example) and the neutron fluence expected at its location in the vessel.

3.2 SAR Criteria

With respect to the adequacy of the surveillance program, information requested for beltline materials includes the following:

(1) Tensile properties.

(2) Dropweight test and Charpy V test results used to determine RT_{NDT}.

(3) Charpy V test results to determine the upper shelf toughness.

(4) Composition, specifically the copper and nickel content.

(5) Estimated maximum fluence for each beltline material.

(6) List of materials included in the surveillance program, with basis used for their selection.

3.3 Surveillance Test Procedures

Surveillance capsules must be removed and tested at intervals in accordance with Appendix H, 10 CFR Part 50. The proposed removal and test schedule should be included in the Technical Specifications.

3.4 Reporting Criteria

All information used to evaluate results of the tests on surveillance materials, evaluation methods, and results of the evaluation should be submitted with the evaluation report. This should include:

(1) Original properties and compositions of the materials.

(2) Fluence calculations, including original predictions, for both surveillance specimens and vessel wall.

(3) Test results on surveillance specimens.

(4) Basis for evaluation of changes in RT_{NDT} and upper shelf toughness.

(5) Updated prediction of vessel properties.

3.5 Technical Specification Changes

Changes in the operating and test limits recommended as a result of evaluating the properties of the surveillance material, together with the basis for these changes, shall be submitted to the Office of Nuclear Reactor Regulation for approval.

4. Pressurized Thermal Shock (PWR only)

4.1 Pressurized Thermal Shock Requirements

As required by 10 CFR 50.61, the following is a summary of requirements for the PWR reactor vessels:

(1) RT_{PTS} values must be projected using end-of-life fluence for each weld, plate or forging in the reactor vessel beltline region. The projected EOL RT_{PTS} values must be approved by the NRC.

(2) PTS screening criteria is 132 °C (270 °F) for plates, forgings, and axial weld materials, and 149 °C (300 °F) for circumferential weld materials.

(3) If reactor vessel is projected to exceed the PTS screening criteria, 10 CFR 50.61(b)(3) requires the applicant to implement a flux reduction program that is reasonably practicable to avoid exceeding the PTS screening criteria.

(4) If the flux reduction program does not prevent the reactor vessel from exceeding the PTS screening criterion at the end of life, the applicant choose between the two options in 10 CFR 50.61 to meet PTS requirements: (a) submit a safety analysis pursuant to 10 CFR 50.61(b)(4) to determine what, if, any, modifications to equipment, systems, and plant operation to prevent failure of the reactor vessel from a postulated PTS event, (b) perform a thermal-annealing treatment of the reactor vessel pursuant 10 CFR 50.61(b)(7) to recover fracture toughness. 10 CFR 50.61 requires details of the approach selected to be submitted for NRC approval at least 3 years before the reactor vessel is projected to exceed the PTS screening criteria.

C. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criteria for Nuclear Power Plants.

2. 10 CFR Part 50, Appendix G, Fracture Toughness Requirements.

3. 10 CFR Part 50, Appendix H, Reactor Vessel Material Surveillance Program Requirements.

4. 10 CFR 50.61, Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events.

5. NRC Memorandum from David Rudland to John Lubinski, Joseph Giitter, and George Wilson, Closure Memorandum Supporting the Limited Revision of NUREG-0800 Branch Technical Position 5-3, “Fracture Toughness Requirements,” April 4, 2017 (ML16364A285).

6. NUREG-0744, Pressure Vessel Material Fracture Toughness.

7. ASME Boiler and Pressure Vessel Code, Section III, including Appendix G, Protection Against Nonductile Failure, American Society of Mechanical Engineers.

8. ASTM E-185-82, Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels.

9. ORNL/NRC/LTR-03/03, Tabulation of Thermally-Induced Stress Intensity Factors (K_{IT}) and Crack Tip Temperatures for Generating P-T Curves per ASME Section XI - Appendix G.

PAPERWORK REDUCTION ACT STATEMENT

The information collections contained in the Standard Review Plan are covered by the requirements of 10 CFR Part 50 and 10 CFR Part 52, and were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0151.

PUBLIC PROTECTION NOTIFICATION

The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.