Technical Letter Report
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Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels

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ABSTRACT

This Technical Letter Report (TLR) is one in a series of reports developed by the Nuclear Regulatory Commission's (NRC's) Component Integrity Branch (CIB) in the Division of Engineering (DE) of the Office of Nuclear Regulatory Research (RES). The series of reports document research efforts associated with reactor pressure vessel (RPV) materials and integrity issues performed by CIB since 2007. The intent of these reports is to document the results of CIB's research efforts for knowledge management purposes, as well as to provide a technical reference for potential future regulatory needs. The current list of topics to be covered by this set of RPV reports is as follows:

A. Methods to Account for the Effect of Embrittlement on the Value of Adjusted Reference Temperature
B. Description of the PFM model used to assess cooldown, heatup, and leak tests
C. Postulated flaw populations used to assess cooldown, heatup, and leak tests
D. Evaluation of the Beltline Region for Nuclear Reactor Pressure Vessels
E. PFM Analyses of Cooldown Transients for Normal Operation
F. PFM Analyses of Heatup Transients for Normal Operation
G. PFM Analyses of Leak Test Transients for Normal Operation
H. Risk informed limits for normal operation
I. Limits for Charpy V-notch energy on the upper shelf
J. Analysis of and toughness requirements for the flange region of the RPV
K. Analysis of and toughness requirements for the nozzle region of the RPV
L. The Effect of Shallow Internal Surface-Breaking Flaws on the Probability of Brittle Fracture of Reactor Vessels Subjected to Normal Cool-Down Transients
M. Tabulation of the results of PFM analyses for normal operation

This report describes the evaluations performed to evaluate the beltline region of the RPV (Item D of the above list).

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ABBREVIATIONS

Abbreviation	Definition
ACRS	Advisory Committee on Reactor Safeguards
ART	Adjusted Reference Temperature
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BAF	Bottom of Active Fuel
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CIB	Component Integrity Branch
Cu	Copper Content
DE	Division of Engineering
EFPY	Effective Full Power Year
EPU	Extended Power Uprate
EOL	End-of-License
ETC	Embrittlement Trend Curve
ft-lbs	Foot-Pounds
LRA	License Renewal Application
LWR	Light Water Reactor
MeV	Million Electron Volts
nvt	Neutron Volt Thermal Energy
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NRO	Office of New Reactors
NSSS	Nuclear Steam Supply System
P-T	Pressure-Temperature
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
RCPB	Reactor Coolant Pressure Boundary
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RPV	Reactor Pressure Vessel
RVID2	Reactor Vessel Integrity Database Version 2.0.1
TAF	Top of Active Fuel
TLR	Technical Letter Report

SYMBOLS

Symbol	Definition
RT_{NDT}	Reference temperature of nil ductility transition for the unirradiated material (°F)
ΔT_{30}, ΔRT_{NDT}	Mean value of the shift in reference temperature at a Charpy V-notch impact energy of 30 ft-lbs (41J) caused by neutron radiation embrittlement (°F)

1.0 OBJECTIVE

This report provides an evaluation of the beltline region of the reactor pressure vessel (RPV). Historically, the "beltline" region of the RPV has been described in both Nuclear Regulatory Commission (NRC) regulations and industry standards (to be fully discussed in Section 2.0) as the region adjacent to the reactor core that must be evaluated to account for the effects of radiation on fracture toughness. For RPV materials, radiation is measured by neutron fluence[†].

As the U.S. fleet of operating reactors ages and continues in service, the beltine region of the RPV becomes larger in extent for several reasons. These reasons include longer operating time periods and reactor core changes (i.e., high burn-up fuel, extended power uprate (EPU) operation, longer fuel bundles, etc.), which lead to greater neutron fluence accumulation (in both amount and area) throughout the RPV beltline and adjacent regions.

The NRC has used a neutron fluence value of $1x10^{17}$ neutrons/centimer^2 (n/cm^2) with damage spectrum energies greater than one million electron volts (E > 1 MeV) at the end-of-license (EOL) to define the extent of the RPV beltline. This definition is described in Section 2.2, Reactor Vessel Design, of NUREG-1511 [1], which ties the definition directly to Title 10, Part 50 to the *Code of Federal Regulations* (10 CFR 50), Appendix (App.) G, "Fracture Toughness Requirements" [2] and App. H, "Reactor Vessel Material Surveillance Program Requirements" [3]:

> *The beltline of the reactor vessel is defined in Appendix G, 10 CFR Part 50, as the region of the reactor vessel that directly surrounds the effective height of the active core and the adjacent regions of the reactor vessel that are predicted to experience sufficient neutron damage to be considered in the selection of the limiting material with regard to radiation damage. The NRC staff considered materials with a projected neutron fluence of greater than 1.0E17 neutrons per square centimeter (n/cm^2) at end of license (EOL) to experience sufficient neutron damage to be included in the beltline. This neutron fluence is based on the surveillance requirements in Appendix H, 10 CFR Part 50....*

In this report, evaluations are documented that re-evaluate the definition of beltline stated in NUREG-1511 based on up to-date operating experience and RPV material surveillance data.

[†] An introductory definition of neutron fluence is provided in Appendix A.

2.0 BACKGROUND

The commercial U.S. light water reactor (LWR) nuclear power industry has identified the need to assess different regions of nuclear RPVs for several reasons. In particular, the RPV beltline region is of interest because of the detrimental effect of neutron exposure on material properties in that region. Therefore, identifying a beltline region of the RPV is a means by which the need to estimate neutron fluence can be limited to just those portions of the RPV where radiation effects are significant, thereby avoiding unnecessary, difficult, or costly evaluations. Such reasoning assumes that there is a region of the RPV exposed to neutron fluence levels below which radiation effects become insignificant. As a consequence, traditionally, the term beltline has been used to define the region immediately adjacent to the active core where significant radiation occurs. The background for the current definition of beltline used in industry standards and regulatory documents is provided in Appendix B. That background is summarized in the remainder of this section based on passages from documents that, collectively, provide the definition of the RPV beltline region to support the context of the evaluations performed in this report.

The definition of beltline from Section 4.3 of ASTM International, formerly known as the American Society for Testing and Materials, E 185-82 (ASTM E185-82) [4], which is the version of this standard currently referenced in 10 CFR 50 App. H [3], is as follows:

> "…*the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core, and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material*."

The definition of beltline from Section 3.1.2 of the most recent version of ASTM E 185-10 [5], is as follows:

> *"...the irradiated region of the reactor vessel (shell material including weld seams and plates or forgings) that directly surrounds the effective height of the active core. Note that materials in regions adjacent to the beltline may sustain sufficient neutron damage to warrant consideration in the selection of surveillance materials."*

In the above quote, regions adjacent to the beltline that may sustain sufficient neutron damage are not considered as part of the beltline. They are only used for consideration in the selection of surveillance materials. This is different from the beltline region defined in 10 CFR 50 App. G [2] or 10 CFR 50.61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events" [6]:

> *"Reactor Vessel Beltline means the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

NRC's definition of the beltline region clearly includes adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage. Both NRC's and ASTM's definitions include terms such as "effective height," "adjacent," and "sufficient neutron damage" to specifically capture beltline effects that extend beyond the active height of the core. As identified in Section 1.0, NUREG-1511 identified a neutron fluence value that defines sufficient

neutron radiation damage to avoid inconsistencies in the development and evaluation of RPV surveillance programs in accordance with 10 CFR 50 App. H [3], pressurized thermal shock (PTS) evaluations in accordance with 10 CFR 50.61 [6], and RPV pressure-temperature (P-T) limits developed in accordance with 10 CFR 50 App. G [2] and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, Nonmandatory Appendix G (ASME App. G) [7]. It is also consistent with the definitions included in 10 CFR 50, App. H and ASTM E185-10.

From Section III of 10 CFR 50 App. H [3], one criterion for surveillance programs is:

> *"No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all uncertainties in the measurements, that the peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E > 1 MeV)."*

According to the 10 CFR 50 App. H [3] criterion, a surveillance program is required for reactor vessel with neutron fluence exceeding $1x10^{17}$ n/cm^2 (E > 1 MeV) at the end of the design life; the description in NUREG-1511 [1] reflects this concept that materials with neutron fluence exceeding $1x10^{17}$ n/cm^2 (E > 1 MeV) at EOL are considered as beltline materials.

The NRC's neutron fluence value for the beltline region differed from a value of $1x10^{18}$ n/cm^2 (E > 1.0 MeV) contained in Section 4 (Item B(5)) of Welding Research Council (WRC) Bulletin 175 [8], which is referenced as one of the technical basis documents in ASME App. G (endnote referenced in G-2120), as follows:

> *"It is not feasible to name a specific fluence above which a surveillance program is mandatory. Available data show, however, that if the fluence at the inner wall is less than 10^{18} nvt (> 1 MeV) no significant radiation damage is to be expected. For higher values of end-of-life fluence the omission of a surveillance program should be justified by showing that for the particular lots of base and weld metal being used, the reactor vessel shell will not become more limiting than other parts of the vessel."*

NRC staff now recognize that, as RPVs are licensed to operate for longer time periods and core changes are adopted (i.e., license renewal, high burn-up fuel, EPU operation, longer fuel bundles, etc.), the RPV region evaluated for significant irradiation effects may extend beyond those regions traditionally evaluated as the beltline region. In addition, with the accumulation of additional operating experience gained from RPV materials surveillance programs, additional data are available to re-evaluate the level of neutron fluence where irradiation effects become significant. Therefore, studies are summarized in this report that re-evaluate the neutron fluence level where irradiation effects are significant on the properties of RPV ferritic materials.

3.0 EVALUATIONS PERFORMED

The following evaluations were performed to review RPV regions and the neutron fluence level where radiation effects are significant:

1. **Axial Neutron Fluence Profile Study**: Axial neutron fluence profiles for sample plants (both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs)) were obtained to study neutron fluence variation above and below the top of active fuel (TAF) and bottom of active fuel (BAF) elevations in the RPV, respectively. The purpose of this study was to identify neutron fluence levels for those RPV regions adjacent to the core.
2. **Adjusted Reference Temperature Study**: Sensitivity studies were performed to investigate the effect of neutron fluence on adjusted reference temperature (ART) calculated by the embrittlement trend curves (ETCs) currently adopted in NRC regulations. Two ETCs are currently used by the NRC: (1) the ETC defined in Section C of Regulatory Guide (RG) 1.99, Revision 2 [9], and (2) the ETC defined in Paragraph (g) of 10 CFR 50.61a, the Alternate Pressurized Thermal Shock (PTS) Rule [10]. The purpose of these studies was to identify neutron fluence levels above which these ETCs predict significant degradation of material fracture toughness caused by irradiation.
3. **Power Reactor Surveillance Data Study**: A study of all available power reactor surveillance data collected through approximately 2002 was performed to investigate the impact of neutron fluence on the shift in measured reference temperature, i.e., ΔRT_{NDT} or ΔT_{30}. The purpose of this study was to review the data from the operating fleet of reactors to see what level of neutron fluence caused a measurable impact on material properties caused by irradiation.

Each of these studies is described in the following sections.

3.1 Axial Neutron Fluence Profile Study

In the beginning of the commercial U.S. power industry, TAF and BAF elevations were used by some of the nuclear steam supply system (NSSS) designers to define the RPV beltline region based on the active height of the nuclear core contained inside the RPV. This definition was consistent with Atomic Energy Commission (AEC) regulatory criteria. For example, the AEC's 1967 Supplementary Regulatory Criteria for ASME Code Constructed Nuclear Pressure Vessels, §1.38, "Attachments to Reactor Vessels" [11] stated the following:

(a) For reactor vessels of ferritic materials, vessel nozzles shall not be located in any shell sections which directly surround the reactor core region and which is calculated to receive integrated neutron doses in excess of $1x10^{17}$ nvt (E of 1 MeV and above).

Therefore, it was common design practice for the NSSS designers to avoid placing stress discontinuities such as nozzles within the active core height region. This design practice also prevented creation of overly-restrictive RPV P-T operating limits caused by the high stress concentration factor and significant radiation effects associated with stress discontinuities within the active core height region.

Neutron fluence profiles that show neutron fluence as a function of RPV elevation are provided in Figure 1 and Figure 2 for sample BWR and PWR plants, respectively, for 32, 50, or 54 effective full power years (EFPY) of operation [12, 13]. These profiles are from relatively recent

evaluations that use the NRC-approved methods of RG 1.190 [14]. The location of the active core is identified in each figure as the region between the BAF and TAF lines. The objective of this study is to find out how the neutron fluence attenuates above and below the active core region. From these figures, the following observations are made:

a. Based on review of available neutron fluence reports and discussions with industry personnel, there are no known undue burdens or limitations on licensees for performing neutron fluence calculations to any reasonable neutron fluence level. Most neutron fluence vendors that were consulted are able to calculate neutron fluence in all regions adjacent to the RPV active core for neutron fluence values as low as$1x10^{16}$ n/cm^2 (E > 1.0 MeV) at EOL. In one of the evaluations associated with Figure 1, neutron fluence was calculated as low as $1x10^{15}$ n/cm^2 (E > 1.0 MeV) at EOL.

b. The difference in required RPV surface area needed for typical and low level of neutron fluence measurements is small. For example, comparing a neutron fluence value of $1x10^{16}$ n/cm^2 (E > 1.0 MeV) at EOL to a value of $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL, Figure 1 and Figure 2 both indicate that an additional vertical height of approximately 40" would be required for evaluation. This additional height is not significant compared to the overall area being evaluated.

As noted in Figure 2, the RPV inlet and outlet nozzles may be within the beltline region depending upon the projected EOL neutron fluence. Figure 2 indicates these nozzles may experience irradiation levels above $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL for some PWR designs. Similarly, instrument, inlet, and outlet nozzles for BWRs (or selected other nozzles for later-vintage BWRs) may also be within a beltline region defined as having a neutron fluence that exceeds $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL. Historically, nozzles were considered to be outside of the beltline region by some licensees because they were located above the TAF elevation or below the BAF elevation. In such cases, any corresponding shift in ART and the associated impact on P-T limit curves for nozzles may have been neglected. However, as identified in 10 CFR 50 App. G, the development of P-T limits must consider all ferritic components for the entire RPV, including nozzles, and the effects of neutron radiation must be considered for any materials that are significantly impacted by radiation.

The results of the Axial Neutron Fluence Profile Study support the following observations:

i. The beltline region may extend above the TAF and below the BAF elevations of the RPV depending upon the projected neutron fluence value at EOL used to define the beltline region. (This neutron fluence value is investigated in the other two studies documented in this report.)

ii. Neutron fluence evaluation beyond the elevations of the TAF and BAF to any reasonable neutron fluence level can be accomplished without undue burden to licensees.

iii. Nozzles close to the reactor core elevation may reside within the beltline region, therefore requiring the impact of irradiation to be evaluated.

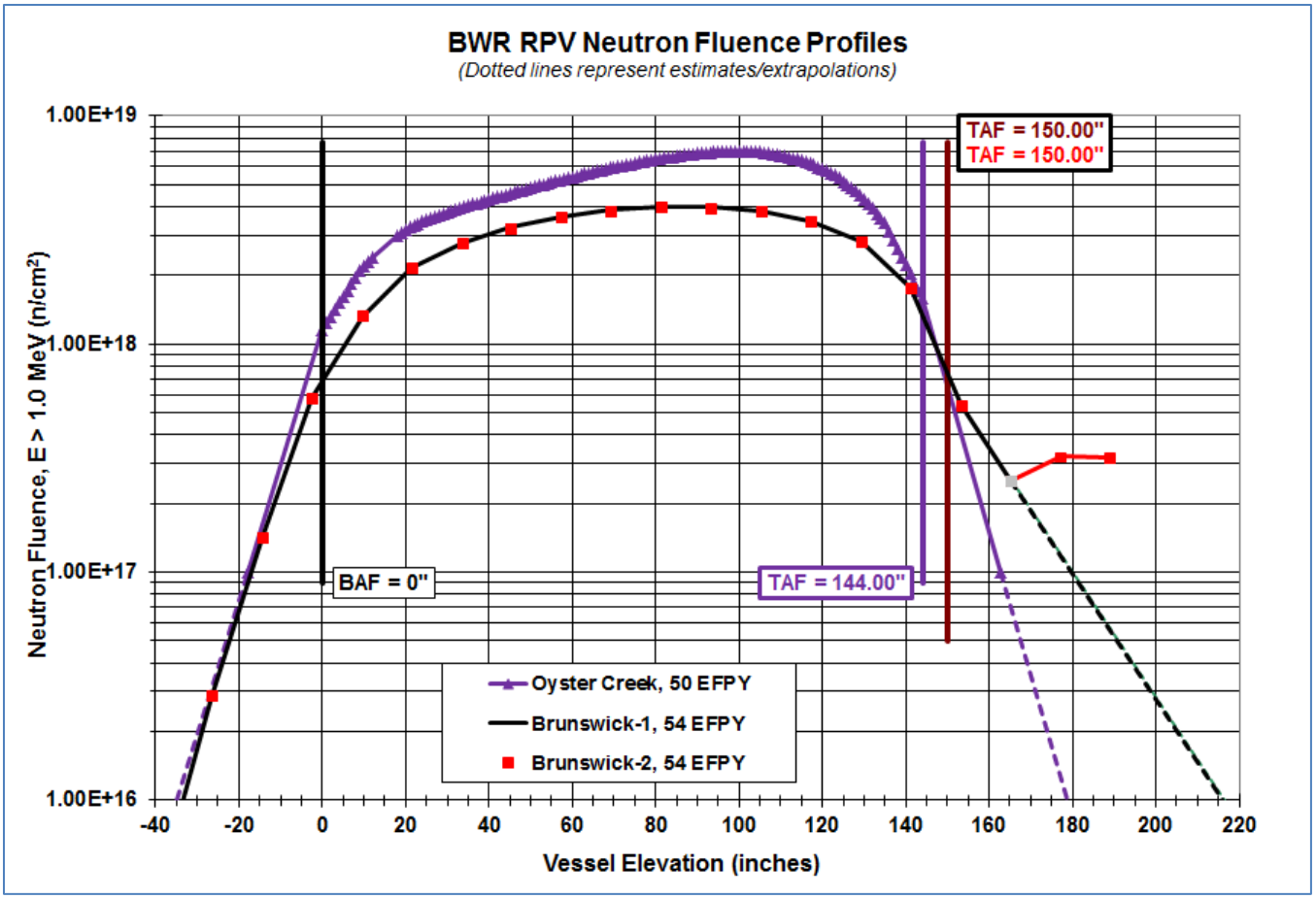

Figure 1. BWR Axial Neutron Fluence Profiles [12]

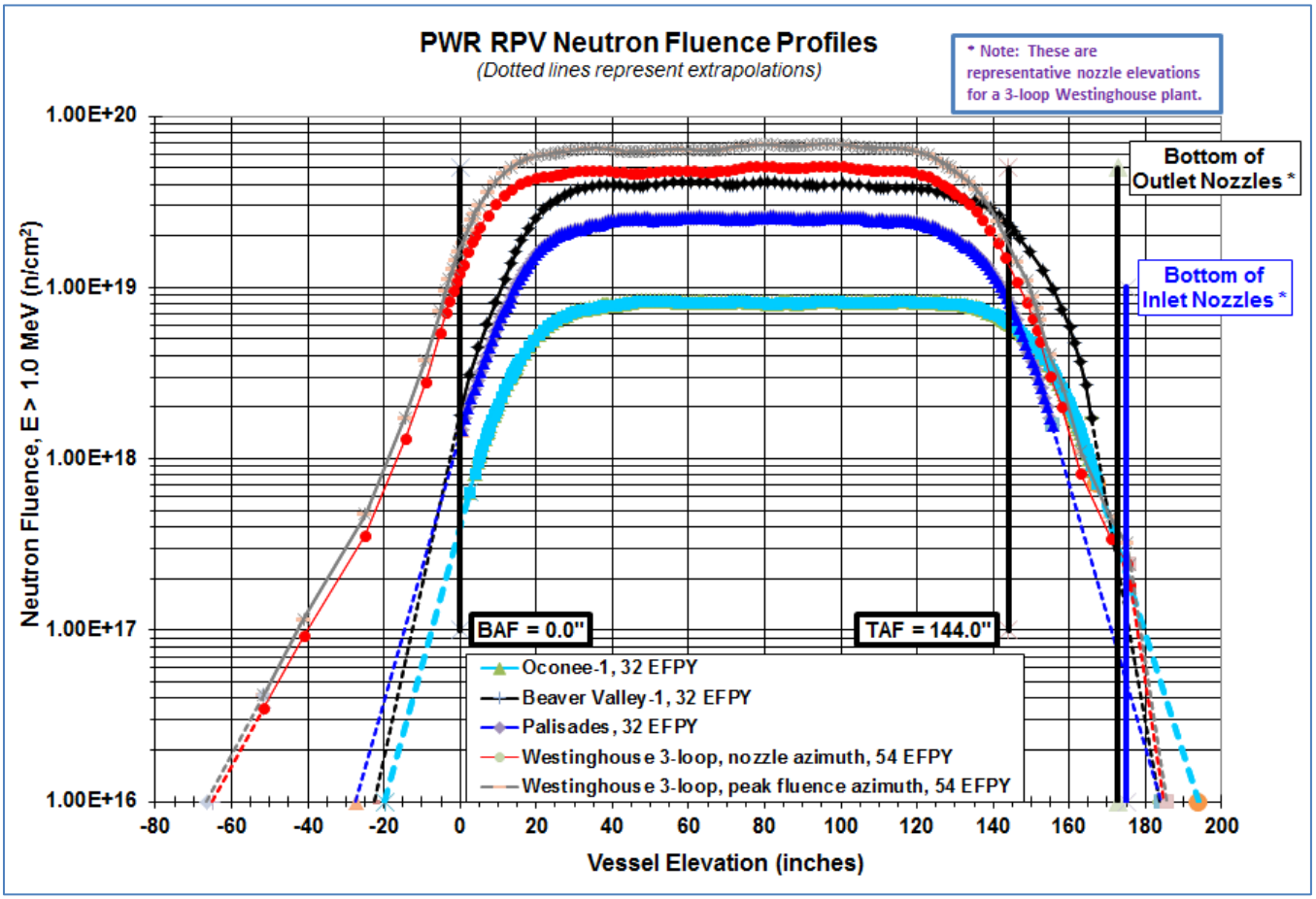

Figure 2. PWR Axial Neutron Fluence Profiles [13]

3.2 Adjusted Reference Temperature Study

For deterministic assessments, adjusted reference temperature (ART) is defined as the initial unirradiated RT_{NDT} plus the shift in the mean reference temperature caused by irradiation plus any applicable margin. ART calculations were performed in this study using the ETCs defined in RG 1.99, Revision 2 [9] and 10 CFR 50.61a [10] to determine whether it is necessary to include embrittlement evaluation for RPV materials having neutron fluence values less than $1x10^{17}$ n/cm^2 (E > 1 MeV) at EOL. RG1.99, Revision 2 and 10 CFR 50.61a define the shift in mean reference temperature caused by irradiation as ΔRT_{NDT} and ΔT_{30}, respectively. The purpose of using both ETCs is to assess the sensitivity of the results due to different ETCs. Both ETCs estimate ART as a function of variables that define the irradiation environment and the chemical composition of the steel.

Information from the NRC's Reactor Vessel Integrity Database Version 2.0.1 (RVID2) [15] was used for the ART calculations. The 40-year EOL (32 EFPY) neutron fluence values in RVID2 were increased by 54/32 to estimate 60-year (54 EFPY) EOL values; therefore, the results of these sensitivity studies reflect the estimated impact of 60 years of plant operation. The results are shown in Table 1 and Figure 3 for the RG 1.99, Revision 2 ETC, and in Table 2 and Figure 4 for the 10 CFR 50.61a ETC. The 381 data points for BWRs and 602 data points for PWRs shown in Table 1 and Table 2 consist of plates, forgings, and welds.

The data in RVID2 summarizes 40-year operating period data for U.S. licensees and, as a result, has not been updated since the 2000 timeframe. Therefore, RES performed an assessment of updates made by licensees to their plant-specific materials data as a part of recent license renewal application (LRA) submittals. The purpose of this assessment was to check the adequacy of the RVID2 data used in the ART Sensitivity Study. This assessment included a sampling of 84 limiting materials reported in NRC-approved LRAs submitted since 1998. For these materials, ΔT_{30} calculations were performed using the 10 CFR 50.61a ETC and the material data from the LRAs. The resulting ΔT_{30} values were compared to ΔT_{30} values resulting from calculations using the 10 CFR 50.61a ETC and the material data from RVID2 for the same 84 materials. The results of this comparison and the related statistics are shown in Figure 5. The diagonal line in the figure represents equal ΔT_{30} values using the two different data sources.

The results of the ART Sensitivity Study support the following observations:

i. For the RG 1.99, Revision 2 ETC:
   a. The average ΔRT_{NDT} remains below 10°F for neutron fluence values less than $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL.
   b. The maximum ΔRT_{NDT} remains below 23°F for neutron fluence values less than$1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL.
ii. For the 10 CFR 50.61a ETC:
   a. For PWRs, the average ΔT_{30} is less than 10°F, and the maximum ΔT_{30} is less than 25°F, for a neutron fluence value equal to $5x10^{16}$ n/cm^2 (E > 1.0 MeV) at EOL.
   b. For BWRs, the average ΔT_{30} is less than 15°F, and the maximum ΔT_{30} is 30°F, for a neutron fluence value equal to $5x10^{16}$ n/cm^2 (E > 1.0 MeV) at EOL.
iii. These results of ΔT_{30} predictions made using RVID2 do not differ significantly from those made using information from LRAs. Therefore, the use of RVID2 for the ART Sensitivity Study, with the results plotted in Figure 3 and Figure 4, is concluded to provide an adequate assessment for the current U.S. operating fleet.

Based on the above observations, it can be concluded that it is unnecessary to include embrittlement evaluation for RPV materials having neutron fluence values less than $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL.

Table 1. ART Sensitivity Study Results for the RG 1.99 Revision 2 ETC

Neutron Fluence: 1.000E+16 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔRT_{NDT} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	1.2	3.4	0.3	0.7	-12.0	53.1	-78.9	28.9
PWR	602	-9.9	91.0	-80.0	35.4	1.0	2.9	0.2	0.7	-7.8	92.4	-79.4	35.2

Neutron Fluence: 2.000E+16 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔRT_{NDT} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	2.3	6.3	0.5	1.2	-9.9	54.2	-78.0	29.1
PWR	602	-9.9	91.0	-80.0	35.4	1.9	5.5	0.4	1.3	-6.0	93.6	-78.9	35.1

Neutron Fluence: 5.000E+16 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔRT_{NDT} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	4.8	13.3	1.0	2.6	-4.9	61.0	-75.9	29.6
PWR	602	-9.9	91.0	-80.0	35.4	4.1	11.7	0.8	2.8	-1.6	96.5	-77.7	34.9

Neutron Fluence: 7.000E+16 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔRT_{NDT} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	6.2	17.1	1.3	3.4	-2.1	64.8	-74.7	29.9
PWR	602	-9.9	91.0	-80.0	35.4	5.3	15.1	1.0	3.6	0.8	98.1	-76.9	34.9

Neutron Fluence: 1.000E+17 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔRT_{NDT} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	8.0	22.2	1.7	4.3	1.5	69.7	-73.1	30.5
PWR	602	-9.9	91.0	-80.0	35.4	6.9	19.7	1.4	4.7	4.0	100.2	-76.0	35.0

Neutron Fluence: 5.000E+17 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔRT_{NDT} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	22.4	62.0	4.8	12.2	30.2	114.1	-60.6	38.4
PWR	602	-9.9	91.0	-80.0	35.4	19.8	56.6	4.0	13.3	29.7	142.8	-68.3	40.3

Neutron Fluence: 1.000E+18 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔRT_{NDT} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	32.7	90.1	6.9	17.8	47.4	139.4	-51.7	42.7
PWR	602	-9.9	91.0	-80.0	35.4	29.1	83.2	5.9	19.5	46.2	159.2	-62.6	45.7

Table 2. ART Sensitivity Study Results for the 10 CFR 50.61a ETC

Neutron Fluence: 1.000E+16 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔT_{30} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	5.5	9.0	3.6	1.0	-3.5	61.3	-71.3	29.3
PWR	602	-9.9	91.0	-80.0	35.4	3.2	6.4	1.6	1.0	-3.5	96.4	-75.9	35.2

Neutron Fluence: 2.000E+16 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔT_{30} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	7.7	14.8	4.6	2.0	0.9	65.2	-68.8	29.7
PWR	602	-9.9	91.0	-80.0	35.4	4.7	11.0	2.1	1.9	-0.5	98.3	-74.7	35.1

Neutron Fluence: 5.000E+16 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔT_{30} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	12.5	29.7	6.5	4.6	10.6	87.2	-64.2	31.3
PWR	602	-9.9	91.0	-80.0	35.4	8.2	23.3	3.0	4.6	6.5	103.9	-72.6	35.2

Neutron Fluence: 7.000E+16 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔT_{30} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	15.1	38.3	7.3	6.2	15.9	99.7	-62.1	32.7
PWR	602	-9.9	91.0	-80.0	35.4	10.2	30.7	3.4	6.2	10.5	112.5	-71.6	35.6

Neutron Fluence: 1.000E+17 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔT_{30} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	18.6	50.0	8.4	8.4	22.7	116.4	-59.6	35.0
PWR	602	-9.9	91.0	-80.0	35.4	12.9	40.8	3.9	8.4	15.9	124.3	-70.4	36.7

Neutron Fluence: 5.000E+17 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔT_{30} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	43.9	134.0	15.2	25.1	64.8	178.8	-43.0	49.0
PWR	602	-9.9	91.0	-80.0	35.4	33.9	118.2	7.0	26.5	52.1	185.7	-62.6	51.6

Neutron Fluence: 1.000E+18 n/cm^2 (E > 1.0 MeV)

	#	Initial RT_{NDT} (°F)				ΔT_{30} (°F)				ART (°F)			
Plant Type	Points	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.	Average	Maximum	Minimum	Std. Dev.
BWR	381	-14.4	52.0	-80.0	28.8	58.5	174.1	19.6	33.8	83.2	206.7	-32.2	53.6
PWR	602	-9.9	91.0	-80.0	35.4	46.1	169.2	9.1	36.7	68.3	208.3	-57.5	58.8

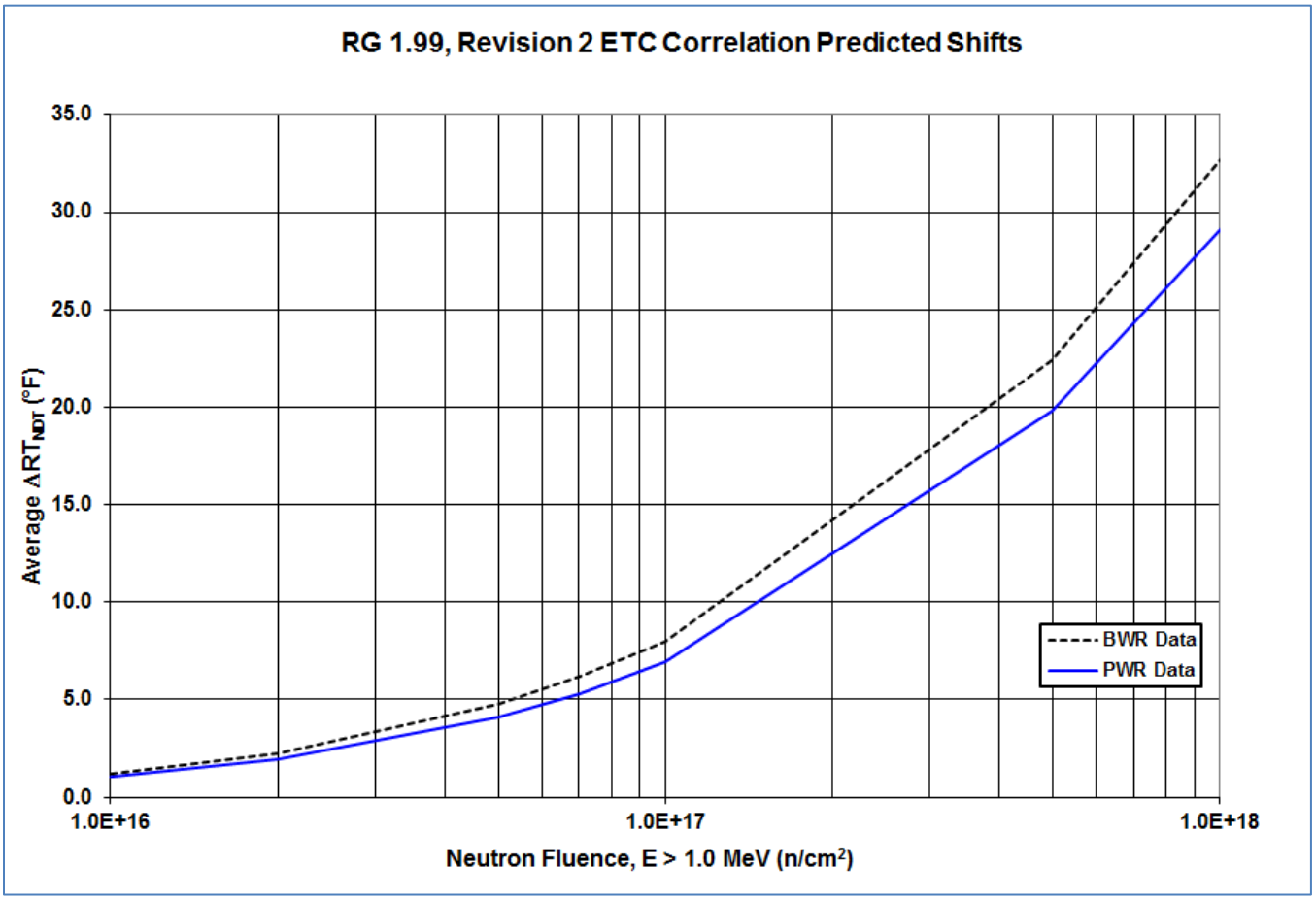

Figure 3. Plot of ART Sensitivity Study Results for the RG 1.99, Revision 2 ETC

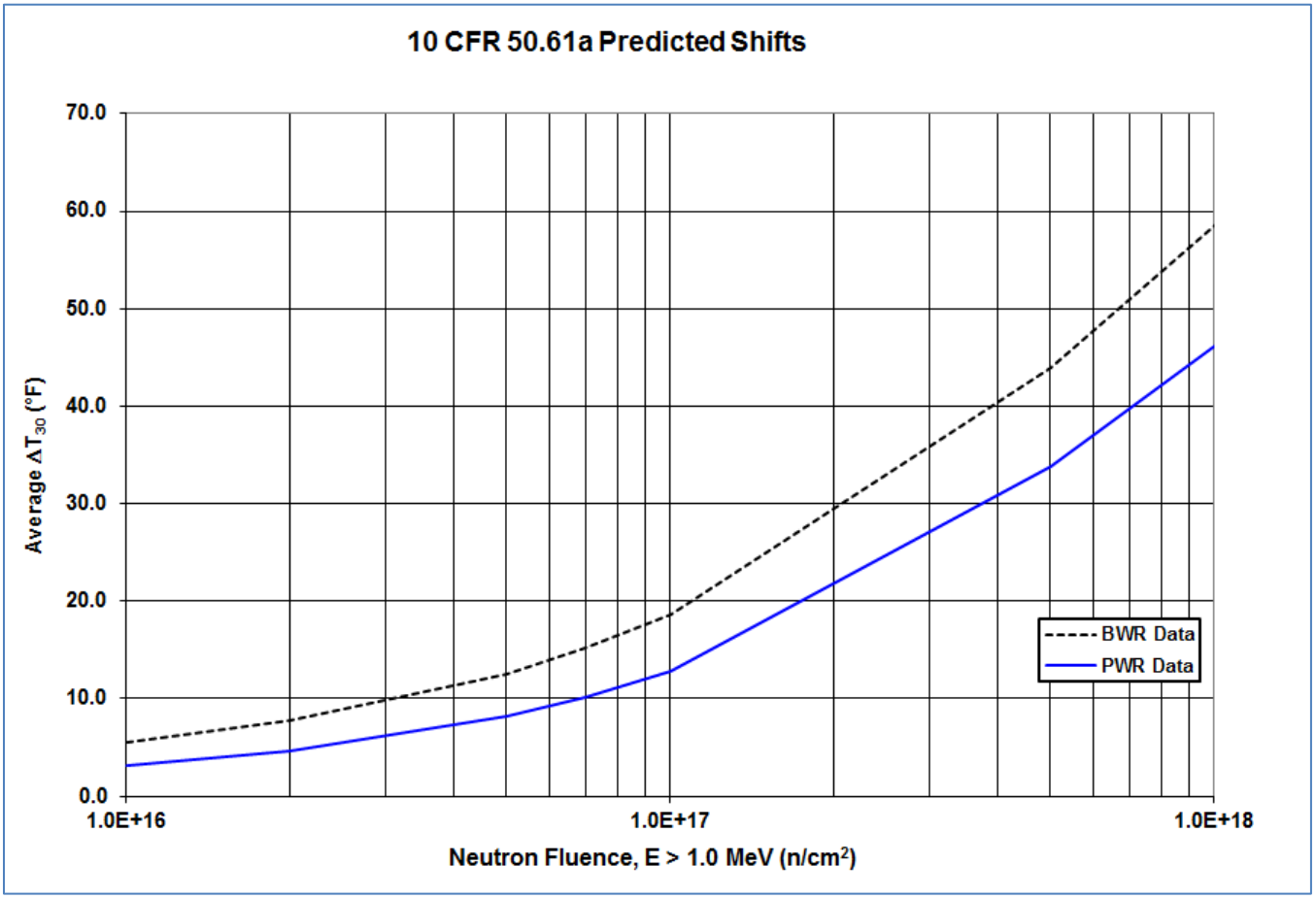

Figure 4. Plot of ART Sensitivity Study Results for the 10 CFR 50.61a ETC

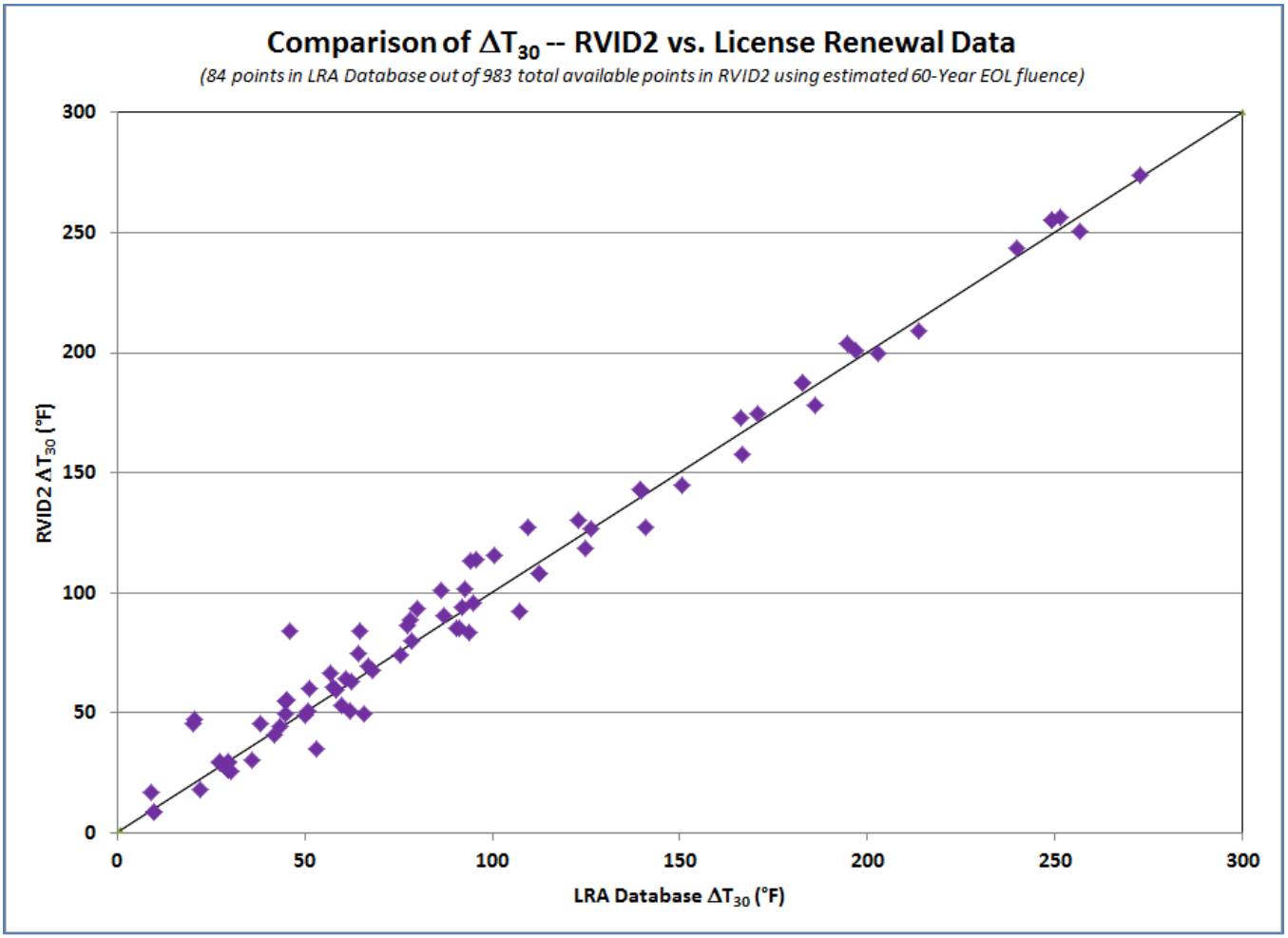

Figure 5. Comparison of 10 CFR 50.61a ETC ΔT_{30} Estimates Made Using RVID2 vs. NRC-Approved License Renewal Materials Data

3.3 Power Reactor Surveillance Data Study

All available surveillance data (prior to 2002) from the database compiled by Eason [16] were reviewed. These results are plotted in Figure 6 in terms of measured ΔT_{30} versus neutron fluence, along with a mean trend line for the data.

The standard deviation (1σ) from the mean of all available ΔT_{30} data in the power reactor database that have a neutron fluence value less than $2x10^{17}$ n/cm^2 (E > 1.0 MeV) (29 points) is 23°F.

The results of the Power Reactor Surveillance Data Study support the following observations:

i. The mean value of ΔT_{30} is greater than 0°F for neutron fluence values above $8x10^{16}$ n/cm^2 (E > 1.0 MeV) at EOL.
ii. The 1σ value of 23°F from the mean trend line is exceeded at a neutron fluence value of approximately $4x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL.

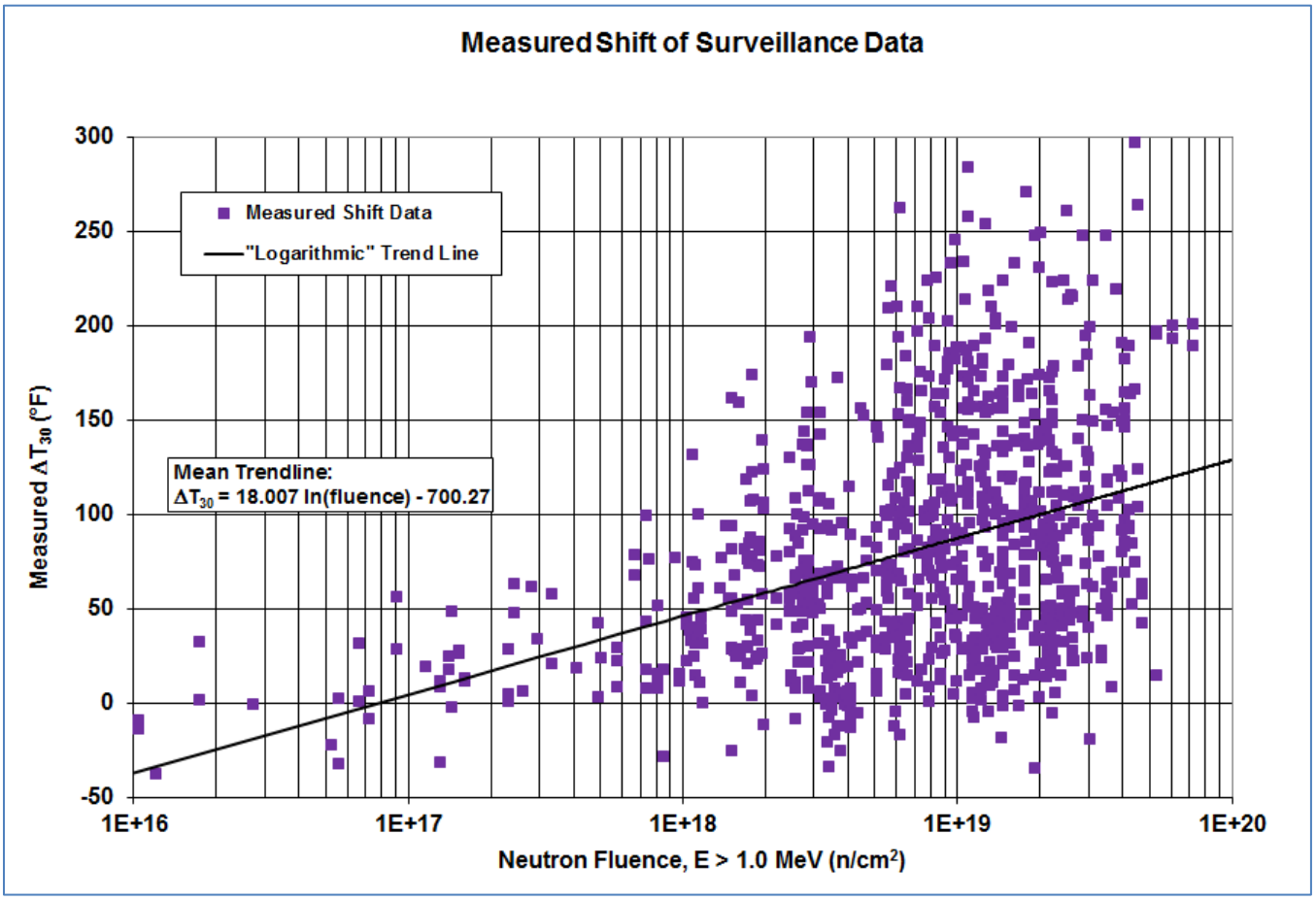

Figure 6. Measured ΔT_{30} vs. Neutron Fluence from Power Reactor Surveillance Data

4.0 SUMMARY OF RESULTS

The results of the three studies summarized in this report indicate that the $1x10^{17}$ n/cm^2 (E > 1.0 MeV) neutron fluence at EOL used by the NRC to define the RPV beltline region where radiation effects are significant remains adequate. Therefore, this value, which appears in NUREG-1511 as a beltline definition and in 10 CFR 50 App. H as a criteria indicating what materials do not need to be included in surveillance programs, is appropriate for continued use.

The foregoing conclusion is based on the following observations from the studies summarized in this report:

i. From the Axial Neutron Fluence Profile Study, neutron fluence levels of $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL may extend approximately 40 inches above and below the RPV active core height. Based on a review of available neutron fluence analyses and discussion with industry personnel, there are not any limitations from a neutron fluence evaluation perspective associated with extending the RPV beltline region by 40 inches or more above and below the region adjacent to the active core necessary to capture all neutron fluence levels to as low as $5x10^{16}$ n/cm^2 (E > 1.0 MeV) at EOL.

ii. From the ART Sensitivity Study, the predicted impact of radiation effects on RPV materials exceeded the scatter inherent to ΔT_{30} surveillance data beginning with neutron fluence values of approximately $1x10^{17}$ n/cm^2 (E > 1.0 MeV) using the RG 1.99, Revision 2 ETC, and approximately $5x10^{16}$ n/cm^2 (E > 1.0 MeV) using the 10 CFR 50.61a ETC.

iii. From the Power Reactor Surveillance Data Study, a measurable impact of radiation effects on RPV materials exceeded the scatter inherent to ΔT_{30} surveillance data at a neutron fluence value of approximately $4x10^{17}$ n/cm^2 (E > 1.0 MeV).

Based on the results of the studies documented in this report, the RES staff recommends the following definition for the RPV beltline region:

> *The beltline is defined as the region of the RPV adjacent to the reactor core that is projected to receive a neutron fluence level of $1x10^{17}$ n/cm^2 (E > 1.0 MeV) or higher at the end of the licensed operating period.*
>
> *Embrittlement effects may be neglected for any region of the RPV if either of the following conditions are met: (1) neutron fluence is less than $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL, or (2) the mean value of ΔT_{30} estimated using an ETC acceptable to the staff is less than 25°F at EOL. The estimate of ΔT_{30} at EOL shall be made using best-estimate chemistry values.*

5.0 REFERENCES

1. NUREG-1511, "Reactor Pressure Vessel Status Report," U.S. Nuclear Regulatory Commission, December 1994, ADAMS Accession No. ML082030506.
2. *Code of Federal Regulations*, Title 10, Energy, Part 50, "Domestic and Licensing of Production and Utilization Facilities," Appendix G, "Fracture Toughness Requirements."
3. *Code of Federal Regulations*, Title 10, Energy, Part 50, "Domestic and Licensing of Production and Utilization Facilities," Appendix H, "Reactor Vessel Material Surveillance Program Requirements."
4. ASTM E 185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM International, approved July 1, 1982.
5. ASTM E 185-10, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels," ASTM International, approved March 1, 2010.
6. *Code of Federal Regulations*, Title 10, Energy, Part 50, "Domestic and Licensing of Production and Utilization Facilities," Paragraph 61, "Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events."
7. ASME Boiler & Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, Nonmandatory Appendix G, "Fracture Toughness Criteria for Protection Against Failure," latest edition approved in 10 CFR 50.55a.
8. Welding Research Council Bulletin 175, "PVRC Recommendations on Toughness Requirements for Ferritic Materials," PVRC Ad Hoc Group on Toughness Requirements, August 1972.
9. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, May 1988.
10. *Code of Federal Regulations*, Title 10, Energy, Part 50, "Domestic and Licensing of Production and Utilization Facilities," §50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."
11. Letter from Harold L. Price (Director of Regulation, Atomic Energy Commission) to Mr. John T. Conway (Joint Committee on Atomic Energy, Congress of the United States), "Supplementary Regulatory Criteria for ASME Code-Constructed Nuclear Pressure Vessels," August 16, 1967, ADAMS Accession No. ML13343A005.
12. BWR Neutron Fluence Data:
    a. Oyster Creek Reactor Pressure Vessel Fluence Profiles, April 5, 2010, ADAMS Accession No. ML101040789.
    b. Brunswick 1&2 - Calculated Neutron Exposure of Unit 1 and Unit 2 Core Shroud/Pressure Vessel, June 26, 2002, ADAMS Accession No. ML021890151.
13. PWR Neutron Fluence Data:
    a. Westinghouse 3-Loop Plant Reactor Pressure Vessel Neutron Fluence Profiles, April 14, 2010, ADAMS Accession No. ML101040818.
    b. Oconee 1, Beaver Valley 1 and Palisades Neutron Fluence Profiles, as described in Section 7.6 of NUREG-1806, Volume 1, "Technical Basis for Revision Of the

Pressurized Thermal Shock (PTS) Screening Limit in the PTS Rule (10 CFR 50.61): Summary Report," May 24, 2006, ADAMS Accession No. ML061580318.

14. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, March 2001.

15. "Reactor Vessel Integrity Database Version 2.0.1," U.S. Nuclear Regulatory Commission, 7/6/2000, http://www.nrc.gov/reactors/operating/ops-experience/reactor-vessel-integrity/database-overview.html.

16. Oak Ridge National Laboratory Report ORNL/TM-2006/530, Eason, E.D., Odette, G.R., Nanstad, R.K., and Yamamoto, T., "A Physically Based Correlation of Irradiation-Induced Transition Temperature Shifts for RPV Steels," November 2007, ADAMS Accession No. ML081000630.

APPENDIX A: Definition of Neutron Fluence

Neutron fluence is flux (either particle or irradiative flux) integrated over time. For particles (such as neutrons), neutron fluence is defined as the total number of particles (neutrons) that intersect a unit area in a specific time interval. Neutron fluence is considered one of the fundamental units in dosimetry. Expressions or quantifications of neutron fluence may be limited to a particular neutron energy range.

Table A-1 provides a list of neutron energy ranges. While the "damage spectrum" ranges from 0.1 MeV to 15 MeV, typically neutron fluence from neutrons above 1 MeV (i.e., fast neutrons) is considered in light water reactor (LWR) material evaluations. These higher energy neutrons have been identified to correlate well with the increase in fracture toughness transition temperature, as identified in NRC Regulatory Guide (RG) 1.190 [A-1], which identifies the "damage spectrum" as 0.1 MeV < E < 15 MeV, noting, "*…the E > 1 MeV fluence has been selected as the exposure parameter for the* RT_{NDT} *and* RT_{PTS} *correlations…*" The NRC requires that a fluence methodology must include dosimetry qualification using good spectral coverage. RPV surveillance capsules contain dosimetry wires of various materials, including iron, nickel, and copper. Recommended threshold detectors are defined in Table 2 of RG 1.190 that allow for sufficient dosimetry spectral coverage.

A.1 References

A-1. Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," U.S. Nuclear Regulatory Commission, March 2001.

Table A-1. Neutron Energy Range Names

Energy Range	Name
0.0 eV to 0.025 eV	Cold neutrons
0.025 eV	Thermal neutrons
0.025 eV to 0.4 eV	Epithermal neutrons
0.4 eV to 0.6 eV	Cadmium neutrons
0.6 eV to 1 eV	EpiCadmium neutrons
1 eV to 10 eV	Slow neutrons
10 eV to 300 eV	Resonance neutrons
300 eV to 1 MeV	Intermediate neutrons
1 MeV to 20 MeV	Fast neutrons
Greater than 20 MeV	Relativistic neutrons

Source: Carron, N.J. *An Introduction to the Passage of Energetic Particles Through Matter*, p. 308, Taylor & Francis 2007. (Wikipedia, http://en.wikipedia.org/wiki/Neutron_temperature).

APPENDIX B: A Historical Perspective for Defining Beltline

Records associated with reactor pressure vessel (RPV) fracture toughness requirements were reviewed to provide a historical perspective on definitions used for beltline and their bases. The results of that review are documented in this Appendix.

In August 1967, the Atomic Energy Commission (AEC) developed "Tentative Regulatory Supplementary Criteria for ASME Code-Constructed Nuclear Pressure Vessels" [B-1]. These criteria did not adopt a value for allowable transition temperature shift (ΔT_{30}), but several sections of that document defined criteria for limiting the impact of irradiation effects, or defining the beltline, as follows:

- §1.20, *Vessel Material Property Improvement*. *For Class A vessels, the material specifications of ferritic materials of any product form (wrought or cast) to be used in the pressure-retaining boundary shall require aluminum killing and vacuum degassing treatment in manufacture, or other treatments producing comparable material property improvement.*
  For reactor vessel ferritic materials which are intended to directly surround the reactor core where the neutron fluence is above 10^{17} nvt‡ (E_n of 1 MeV or above), the material specification shall limit the phosphorus content to 0.012 percent maximum and the sulfur content to 0.015 percent maximum for both ladle and check analysis.…
- §1.25, *Ductile Brittle Transition Properties*: *(d) For materials directly surrounding the reactor core, including welds and weld heat-affected zones, the upper shelf absorbed energy test value of any longitudinal specimen of carbon and low alloy steels shall be no less than 60 ft.-lbs. at a temperature no higher than 160 F....*
- §1.38, *Attachments to Reactor Vessels*: *(a) For reactor vessels of ferritic materials, vessel nozzles shall not be located in any shell sections which directly surround the reactor core region and which is calculated to receive integrated neutron doses in excess of $1x10^{17}$ nvt (E_n of 1 MeV and above)....*

In April 1970, the Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on NDT/Pressure Vessels/Containment met with the Regulatory Staff and ACRS consultants in Oak Ridge, TN to discuss proposed criteria drafted earlier that year for Appendix F§, "Reactor Material Surveillance Program Requirements" [B-3]. Those criteria did not adopt a limit for the mean value of the shift in reference temperature at an impact energy of 30 ft-lbs, ΔT_{30}, but the beltline was defined in Section II, Definitions, as follows:

> A. *"Beltline region of reactor vessels" means the shell material, including welds and weld heat-affected zones, which directly surround the effective height of the reactor core.*
> B. *"Effective height of reactor core" is not less than the overall height of the reactor fuel element assemblies, and in no case less than the height of vessel internal thermal shields where used.*

In addition, an editorial write-in comment was noted for Item B above that stated, "Reword to accommodate areas above & below core."

‡ "nvt" is an outdated abbreviation given to time-integrated neutron flux, as measured in neutrons per square centimeter.

§ Draft Appendix F was eventually published as 10 CFR 50 Appendix G.

A transmittal of the April 1970 ACRS Subcommittee meeting was sent to the ACRS Chairman from the Division of Reactor Standards on April 24, 1970 [B-4]. That transmittal incorporated comments from the April ACRS Subcommittee meeting, and requested ACRS review and comment on two draft criteria documents that were also considered suitable for publication for comments in the Federal Register: (i) 10 CFR 50.55(i), "Fracture Toughness Criteria," and (ii) Appendix F, "Reactor Material Surveillance Program Requirements." Section (2), "Definitions," of the draft of 10 CFR 50.55(i) contained the following:

> *(viii) "Beltline region of reactor vessel" comprises the shell material, including welds and weld heat-affected zones, which directly surrounds the effective height of the fuel element assemblies, plus any additional material for which the predicted shift of the Charpy V-notch (C_V) fracture energy curve exceeds 100°F.*

Section II, "Definitions," of Appendix F contained the following:

> *A. "Beltline region of reactor vessel" is defined in 10 CFR Part 50,* § 50.55a (i)(2).

Therefore, the ACRS Subcommittee recommended that a ΔT_{30} value above 100°F be required before a material is considered to be in the beltline as a part of the original publication of 10 CFR 50 App. G. The following revised definition was published in Volume 36 of the 1971 Federal Register [B-5], which represented the final draft of 10CFR50 Appendix G released for public comments (see Section II, "Definitions"):

> *E. "Beltline region of reactor vessel" comprises the shell material, including welds and weld heat-affected zones, which directly surrounds the effective height of the fuel element assemblies, and any additional height of shell material for which the predicted shift of the Charpy V-notch (C_V) fracture energy curve exceeds 100°F.*

In June 1973, a Consent Calendar Item (SECY-R 700) [B-6] was entered for the Commissioners through the Director of Regulation that recommended formal publication of amendments to 10 CFR 50 that included the then new Appendix G, "Fracture Toughness Requirements," and Appendix H, "Reactor Vessel Material Surveillance Program Requirements." This transmittal represented a final compilation of all of the proceeding discussions into the first formally-published versions of 10 CFR 50 Appendices G and H. Discussion of modifications to the above definitions was included, and the definitions were revised. For 10 CFR 50 App. G, the definition of "beltline region of the reactor vessel" was broadened to include more shell material above and below the core and the ΔT_{30} requirement was lowered to 50°F, as follows:

> *[~~E~~]H. "Beltline region of reactor vessel" [~~comprises~~] means the shell material (including welds and weld heat-affected zones) which directly surrounds the effective height of the fuel element assemblies and any additional height of shell material for which the predicted [~~shift of the Charpy V-notch (C_V) fracture energy curve~~] adjustment of reference temperature at end of service life of the reactor vessel exceeds [~~100°~~] 50°F.*

For 10 CFR 50 App. H, the implied definition of beltline was recommended in the following form in Section II, "Surveillance Program Criteria":

> *A. No material surveillance program is required for reactor vessels for which it can be conservatively demonstrated by analytical methods, applied to experimental data and tests performed on comparable vessels, making appropriate allowances for all*

> *uncertainties in the measurements, that the peak neutron fluence (E>1MeV) at the end of the design life of the vessel will not exceed [~~$5x10^{16}$~~] 10^{17} n/cm^2.*

Attachment B to the Consent Calendar Item provided analysis of the public comments received on the draft version of the appendices, which included a recommendation that the beltline be defined based on a neutron fluence level of 10^{19} n/cm^2 (E > 1.0 MeV) at EOL. The Staff did not accept this comment.

The above definitions for 10 CFR 50 Appendices G and H were formally published in Volume 38 of the Federal Register in July 1973 [B-7]. The 10 CFR 50 App. G definition established the first formally published criteria associated with defining the beltline that involved ΔT_{30}, and required it to exceed +50°F.

In November 1980, Volume 45 of the Federal Register [B-8] contained proposed amendments to 10 CFR 50 Appendices G and H for public comments. No changes were proposed to the 10 CFR 50 App. H definition of beltline, but changes to the 10 CFR 50 App. G definition were proposed "to more clearly delineate which parts of the vessel are of concern from the standpoint of surveillance of radiation damage" as follows:

> *G. "Beltline" or "Beltline region of reactor vessel" means the region of the reactor vessel (shell material including welds, heat-affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regard to radiation damage."*

These definitions for 10 CFR 50 Appendices G and H were published in May 1983 in Volume 48 of the Federal Register [B-9], and they became effective on July 26, 1983. 10 CFR 50 was published with these changes in January 1984 [B-10]. These same definitions have remained in place until the current day. Since the most recent 10 CFR 50 App. G definition removed all reference to neutron fluence levels or ΔT_{30} values, the definition of the beltline from NUREG-1511 [B-11] and 10 CFR 50 App. H for a neutron fluence level of $1x10^{17}$ n/cm^2 (E > 1.0 MeV) at EOL has been utilized by the NRC staff to establish the level where significant irradiation effects occur.

Figure B-1 graphically summarizes the historical evolution of the definition of beltline as described in this Appendix, including the definition suggested in Section 4.0 of this report.

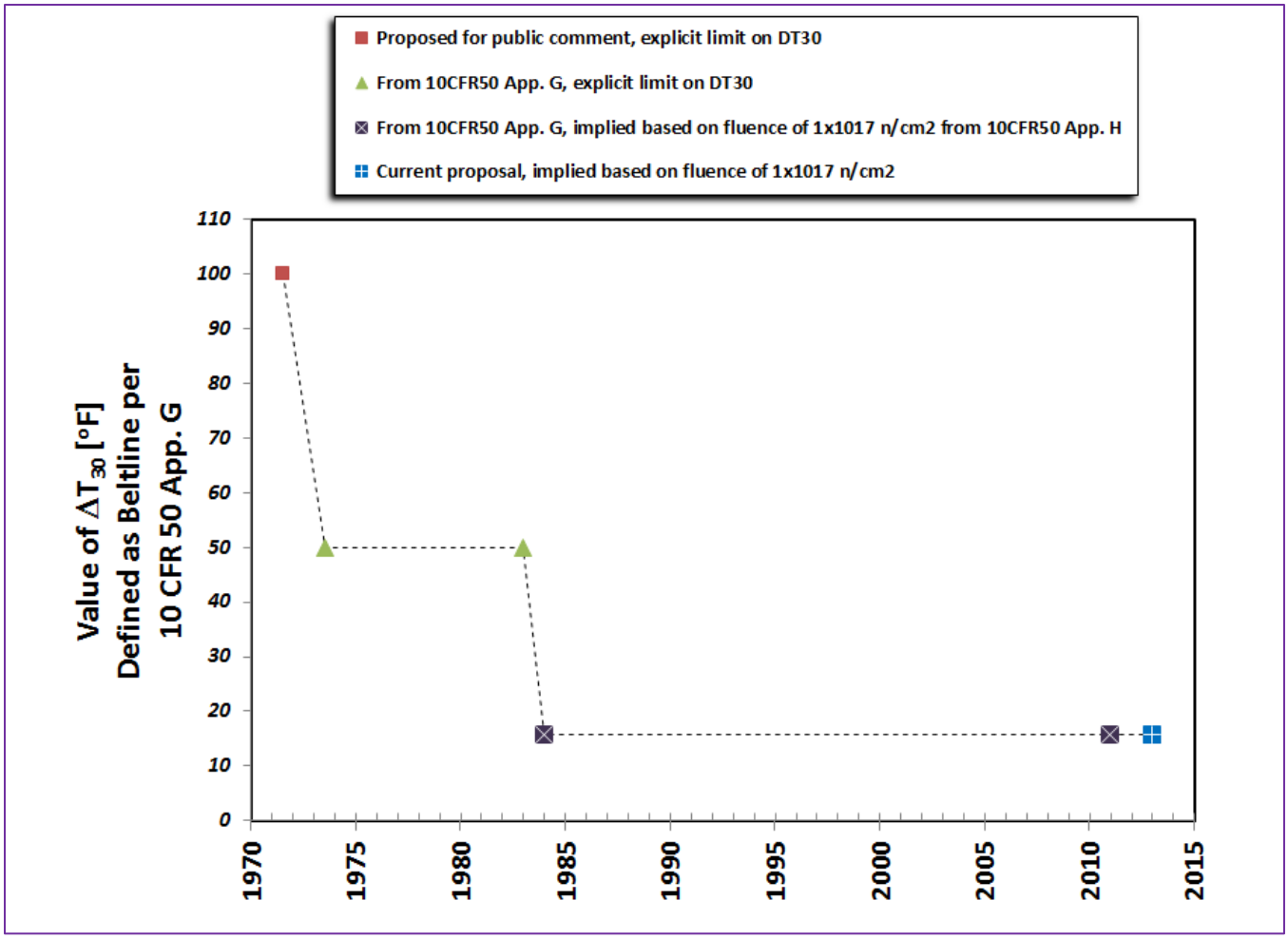

Figure B-1. Historical Variation in ΔT_{30} Associated with Definition of Beltline

B.1 References

B-1. Letter from Harold L. Price (Director of Regulation, Atomic Energy Commission) to Mr. John T. Conway (Joint Committee on Atomic Energy, Congress of the United States), "Supplementary Regulatory Criteria for ASME Code-Constructed Nuclear Pressure Vessels," August 16, 1967, ADAMS Accession No. ML13343A005.

B-2. ACRS-0620, Minutes for the ACRS NDT/Pressure Vessels/Containment Subcommittee Meeting, Washington, DC, January 22, 1970, ADAMS Accession No. ML13343A005.

B-3. Minutes for the ACRS NDT/Pressure Vessels/Containment Subcommittee Meeting, Oak Ridge, TN, April 2 1970, ADAMS Accession No. ML13343A005.

B-4. Letter to Dr. Joseph M. Hendrie (Chairman, ACRS) from Edson G. Case (Director, Division of Reactor Standards), Draft of 10 CFR Part 50.55a(i), "Fracture Toughness Criteria" for publication in the Federal Register, April 24, 1970, ADAMS Accession No. ML13343A005.

B-5. Federal Register, Vol. 36, No. 129, pp. 12697-12700, July 3, 1971, ADAMS Accession No. ML13343A005.

B-6. SECY-R 700, AEC Commissioners Consent Calendar Item, "Amendment to 10 CFR Part 50: Appendix G, 'Fracture Toughness Requirements,' and Appendix H, 'Reactor Vessel Material Surveillance Requirements'," June 1, 1973, ADAMS Accession No. ML13343A005.

B-7. Federal Register, Vol. 38, No. 136, pp. 19012-19016, July 17, 1973, ADAMS Accession No. ML13343A005.

B-8. Federal Register, Vol. 45, No. 222, pp. 75536-75539, November 14, 1980, ADAMS Accession No. ML13343A005.

B-9. Federal Register, Vol. 48, No. 104, pp. 24008-24011, May 27, 1983, ADAMS Accession No. ML13343A005.

B-10. Code of Federal Regulations, Chapter I, Nuclear Regulatory Commission, Appendices G and H, pp. 481-485, 1984, ADAMS Accession No. ML13343A005.

B-11. NUREG-1511, "Reactor Pressure Vessel Status Report," U.S. Nuclear Regulatory Commission, December 1994, ADAMS Accession No. ML082030506.