Supporting Document

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" Update of Irradiation Embrittlement in
BAW-10008 Part 1 Rev. 1" Non-Proprietary

Update of Irradiation Embrittlement in BAW-10008 Part 1 Rev. 1

AREVA NP Inc. Document No. 47-9048125-002

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NOTE:
This document is the non-proprietary version of the AREVA NP Inc. proprietary document 51-9038244-002. This document is identical to 51-9038244-002 except the proprietary information have been deleted as marked by "[ ]".

1 Purpose

This document is the deliverable for the Subtask 2.7b "Update BAW-10008 Part 1" of AREVA NP Inc. Job 4160812, Revision 3[1], "PWR Internals Components Functionality Analysis for the B&W Design" for the Electric Power Research Institute (EPRI).

2 Background and Scope

2.1 BAW-2248A Section 4.5.2, NRC FSER Section 3.4.2

Subtask 2.7b is to address part of the commitment made in BAW-2248A[2] during the Babcock & Wilcox Owners Group (B&WOG) Generic License Renewal Program (GLRP) in the 1990s. BAW-2248 is the B&WOG GLRP topical report for the reactor vessel internals. Section 4.5.2 of BAW-2248A identifies an action item of updating Appendix E to BAW-10008 Part 1, Rev. 1[3] (hereafter referred to as BAW-10008) concerning neutron irradiation embrittlement for the license renewal period. Section 4.5.2 of BAW-2248A is quoted below:

"4.5.2 Ductility – Reduction of Fracture Toughness

BAW-10008, Part 1, Rev. 1, documents the acceptability of the reactor vessel internals under LOCA and a combination of LOCA and seismic loadings. The effect of irradiation on the material properties and deformation limits for the internals is presented in Appendix E where it is concluded that at the end of 40 years, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits.

Existing literature was reviewed with regard to irradiation of the reactor vessel internals. Literature reviewed were: (1) BAW-2060, "Project Topical Report: Oconee Nuclear Power Station Unit 1 Reactor Internals Life Extension Project"; (2) EPRI TR-103838, "PWR Reactor Pressure Vessel Internals License Renewal Industry Report; Revision 1"; and (3) Sixth International Symposium on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors. Two conclusions were drawn from the review:

1) *Additional testing of the physical and mechanical property changes of irradiated material and continued surveillance of the reactor internals should be performed to provide reliable data on the irradiated properties of stainless steel.*

2) *ASME Section XI visual (VT-3) examination standards for category B-N-3 is a current and effective program for detection of cracks and repair/replacement for vessel internals components that are accessible by removal of the core and/or other internals.*

However as noted in Section 4.2, Examination Category B-N-3 may not be adequate to detect reduction of fracture toughness in components. A program is being implemented to manage the effects of aging due to the reduction of fracture toughness of the reactor vessel *internals. This aging management program is discussed in Section 4.6. Hence, this TLAA will be resolved on a plant-specific basis per 10 CFR 54.21 (c)(1)(iii) based on the results and conclusion of the program."*

The final safety evaluation report (FSER) of BAW-2248 by the US Nuclear Regulatory Commission (NRC) was issued on December 9, 1999. Section 3.4.2 of the NRC FSER provided the following evaluation to Section 4.5.2 of BAW-2248:

"3.4.2 Ductility – Reduction of Fracture Toughness

Section 4.5.2 of BAW-2248 describes a TLAA related to the acceptability of the reactor vessel internals under loss-of-coolant-accident (LOCA) and seismic loading. The topical report states that Appendix E to BAW-10008, Part 1, Rev. 1, concludes "that at the end of the 40 years, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits." The topical report indicates that this TLAA will be resolved on a plant-specific basis per 10 CFR 54.21(c)(1)(iii) based on the results and conclusions of the planned RVIAMP.

The aging management approach proposed for neutron irradiation embrittlement (Section 3.3.4) includes examination for cracking as the primary management method for RVI components. This examination is effective in managing neutron embrittlement as it relates to the fracture resistance of the RVI component material in the presence of flaws. The deformation limits and adequate ductility described above in Appendix E to BAW-10008, Part 1, Rev. 1, relates to the material behavior in the absence of flaws. Therefore, no examination looking for flaws will be sufficient to demonstrate resolution of this TLAA, which will require determination of the expected material properties at the end of the license renewal period. As described in the topical report, the planned RVIAMP program will provide the data necessary to resolve this TLAA.

Therefore, this item should be addressed as a renewal applicant action item on a plant-specific basis. This is Renewal Applicant Action Item 12."

Note:
TLAA – time-limited aging analysis
RVIAMP – reactor vessel internals aging management program

2.2 BAW-10008 Part 1, Rev. 1, Appendix E

BAW-10008 used properties of unirradiated Type 304 austenitic stainless steel in determining allowable stresses. Section 3.2 of BAW-10008 stated that *"...In Appendix E – a discussion of the effect of irradiation on material properties – it is concluded that the use of unirradiated properties is conservative."* Appendix E – Effect of Irradiation on the Material Properties and Deformation Limits for Reactor Internals is quoted below.

"All structural materials of the reactor internals (except for bolts) are 304 stainless steel. Curves showing the effect of irradiation on the yield strength, ultimate strength, and ductility of 304 SS are given in Figures E-1, E-2, and E-3. (Note that these are typical properties. Minimum specification properties are used in selecting design allowable stress.) These curves show that the strength of 304 SS is enhanced by irradiation; thus, margins of safety for the occurrence of LOCA and/or earthquake are increased with reactor internals exposure.

The maximum primary stress permitted in the internals for Case IV (LOCA plus earthquake) is limited to two-thirds of the ultimate strength. As irradiation increases the yield strength to this value, plastic strains approach zero, so that the small loss of ductility is inconsequential.

The maximum fluence in the core barrel in the region near flanges is much less than 10^{20} nvt (E > MeV) at the end of a 40-year design lifetime. This is the region of maximum stress intensity and the region where a loss of ductility would be detrimental. As noted in Figure E-3, the uniform elongation is greater than 20% for this fluence. It is concluded that even at the end of a 40-year lifetime, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and the irradiation will not adversely affect deformation limits."

2.3 *Scope*

The internals locations where the loss of ductility due to neutron irradiation was considered detrimental by Appendix E to BAW-10008 are the core barrel flanges. Specifically, Appendix E stated that the maximum fluence near the core barrel flange region would be much less than 10^{20} nvt, E>1 MeV (note, the fluence unit "nvt" is an archaic form of "n/cm^2") at the end of a 40-year design lifetime. Based on Figure 2-1 (Figure E-3 in BAW-10008), the typical uniform elongation would be greater than 20% for this fluence and therefore Appendix E concluded that,

> *"even at the end of a 40-year lifetime, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits."*

The scope of this document is to address the commitment in Section 3.4.2 of the NRC FSER of BAW-2248 related to the above conclusion for a 60-year lifetime. Specifically, the commitment is as follows:

> *"...resolution of this TLAA... will require determination of the expected material properties at the end of the license renewal period."*

Therefore, this document will:

1. update fluence of the core barrel flanges to a 60-year lifetime;
2. examine the validity of Figure E-3 in BAW-10008;
3. examine the "deformation limits" assumed in BAW-10008;
4. update Appendix E to BAW-10008 to a 60-year lifetime.

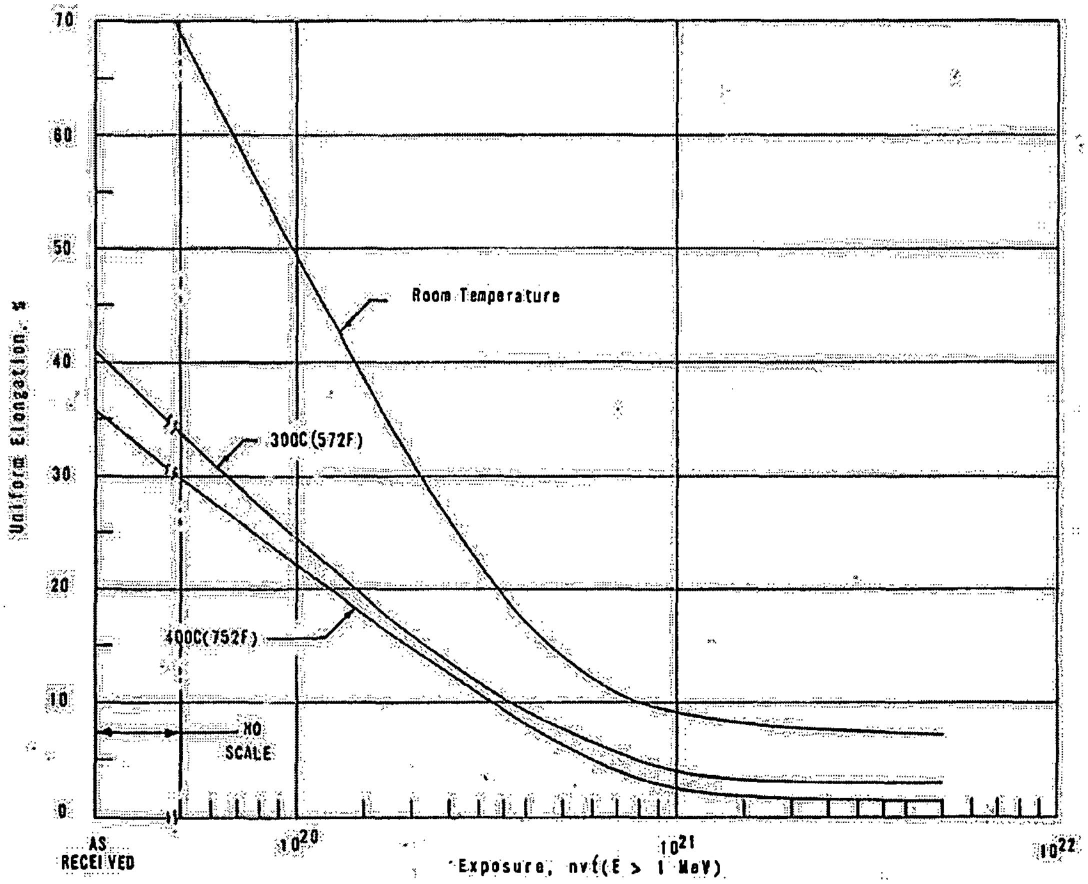

Figure 2-1 Effect of irradiation on uniform elongation of Type 304SA irradiated at 290°C (554°F) and tested at various temperatures. (Note, this is Figure E-3 in Appendix E to BAW 10008 Part 1, Rev. 1[3]).

3 Evaluation

3.1 Locations of Maximum Flux for Core Barrel Flanges

[

] [4]

3.2 Fluence of Core Barrel Flanges

Based on Section 3.1, the maximum neutron flux locations of the core barrel flanges, i.e., locations closest to the core, are the following:

1. Top core barrel flange: [ ] below the top surface of the top flange on the I.D. surface
2. Bottom core barrel flange: I.D. surface [ ] above the bottom surface of the bottom flange on the I.D. surface.

The maximum flux values are listed in Table 3-1. These flux values are based on a hypothetical fuel cycle design that has a conservatively high neutron leakage. The high-leakage comes from loading high powered fuel assemblies on the periphery of the core. The peripheral assembly powers are so high that the leakage represents a bounding value; no actual fuel cycle design would be expected to have a higher neutron flux leaving the core. Thus, the flux values in Table 3-1 represent an upper bound for the core barrel flanges. In addition, the neutronics model represents an approximation of the axial material regions between the core and flange locations. The material approximations allow a higher neutron flux to reach the barrel flanges than would be expected with a more detailed model that contained fewer approximations.

The bounding flux values are transformed to bounding fluence values at the end of a 60-year lifetime by utilizing the following conservative assumptions for the plant's capacity factor:

- [ ]
- [ ]

Based on the above load capacity factor, the 60-year lifetime represents 54 effective full power years (EFPY), or 90% overall load capacity factor over the 60-year lifetime. The 60-year lifetime maximum fluence for the core barrel flanges are listed in Table 3-1.

Table 3-1 Maximum Fluence of the Core Barrel Flanges

Location of Maximum Fluence	Flux, n/cm^2/sec, E > 1 MeV	60-Year (54 EFPY) Fluence, n/cm^2, E > 1 MeV
Top core barrel flange: [ ] below the top surface of the top flange (I.D. surface)	[ ]	[ ]
Bottom core barrel flange: [ ] above the bottom surface of the bottom flange (I.D. surface)	[ ]	[ ]

3.3 *Tensile Data of Type 304SA Irradiated to Moderate Fluence*

The [ ] maximum 60-year fluence for the top core barrel flange remains below the 10^{20} n/cm^2, E>1 MeV assumed in Appendix E for a 40-year lifetime. However, the maximum 60-year fluence for the bottom core barrel flange is [ ], exceeds the 10^{20} n/cm^2, E>1 MeV assumption.

Figure 2-1 shows a rapid decrease in uniform elongation between fluence [ ], a fluence level relevant to the bottom core barrel flange. Although the reference was not listed in BAW-10008, the Figure 2-1 data are most likely from Type 304SA irradiated in fast breeder reactors. In order to examine the validity of Figure 2-1 for the bottom core barrel flange, a search for more recent Type 304SA test data between [ ], especially from BWRs or PWRs, has been conducted. The search results are summarized below.

<u>Decommissioned PWR Core Barrel – MRP-128[5] and MRP-129[6]</u>

MRP-129 reported tensile tests of Type 304SA stainless steels removed from vessel internals of a decommissioned PWR. A specimen machined from the removed Type 304SA core barrel had accumulated [ ]. The irradiation temperature of the removed core barrel sample was between 535 and 575°F (279 and 302°C) over the PWR operating history. This is also similar to the operating temperature of 560°F (293°C) for the bottom core barrel flange. The tensile specimen geometry machined from the removed core barrel is shown in Figure 3-3. The tensile test was performed at a strain rate ~7.6×10^{-4}/sec. The stress-strain curve at 608°F (320°C) is shown in Figure 3-4 and the test results are summarized in Table 3-2.

Table 3-2 Tensile Properties of Type 304SA Core Barrel from a Decommissioned PWR[5, 6]

Irradiation temperature 535 to 575°F (279 to 302°C)

Specimen ID	Test Temp.		Fluence		0.2% Yield	Ultimate Tensile	Uniform Elong.	Total Elong.
	°C	°F	dpa	n/cm^2 E > 1 MeV	MPa	MPa	%	%
[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

BWR Riser Pipe – BWRVIP-35[7]

In 1997, BWRVIP-35 reported tensile test results of stainless steels removed from reactor vessel internals of two Swedish BWRs, [ ]. The tested materials included portions of a riser pipe (18 years in the reactor core) containing Type 304SA base metal from [ ]. The irradiation temperature for the riser pipe is estimated as the core coolant temperature of 280°C (536°F) and the fluence was up to [ ]. The tensile specimen geometry machined from the riper pipe is shown in Figure 3-5. The tensile test results are summarized in Table 3-3.

Table 3-3 Tensile Properties of Type 304SA from BWRs[7]

Irradiation Temperature 280°C (536°F)

BWR Component	Specimen ID	Test Temp.		Fluence	0.2% Yield	Ultimate Tensile	Uniform Elong.	Total Elong.
		°C	°F	n/cm^2 E > 1 MeV	MPa	MPa	%	%
Riser Pipe Base Metal (Type 304 SA)	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]
	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]	[ ]

BWR Neutron Absorber Tube and Control Blade Sheath – Chung et al.[8]

The 1993 paper by Chung et al. reported tensile property data of 304SA neutron absorber tubes and a control blade sheath extracted from several un-named operating BWRs. The slow strain rate test (SSRT) specimen size machined from the control blade sheath was 57.2mm x 12.7mm x 1.22mm. The SSRT stress-strain curves of commercial purity Type 304SA neutron absorber tubes and the control blade sheath irradiated to moderate fluence levels are shown in Figure 3-6. The irradiation temperature of these BWR component items was not reported, but could be assumed to be approximately 288°C (550°F). Results of the SSRT tested in air at 289°C (552°F) are summarized in Table 3-4.

Table 3-4 SSRT Results of Commercial Purity Type 304SA from BWRs[8]

Tested in Air at 289°C (552°F)[(1)], Strain Rate 1.65 x 10^{-7}/sec.

Specimen ID	Hot-Cell ID	Source Heat ID	SSRT No.	Fluence	0.2% Yield	Ultimate Tensile	Uniform Elong.[(2)]	Total Elong.
				n/cm² E > 1 MeV	MPa	MPa	%	%
BL-BWR-2L	389E1A	CP304-A	IR-2	2 x 10^{20}	184	390[(3)]	15.6[(3)]	15.6[(3)]
BL-BWR-2M	389E2D	CP304-A	IR-3	6 x 10^{20}	221	465	30	34.8
C7B1W	LCS-5	CP304-B	IR-17	2.3 x 10^{20}	360	577	33	34.1
C7M2K	LSC-11	CP304-B	IR-23	5.3 x 10^{20}	570	644	17	21.4

1. The irradiation temperature inside the BWRs was not reported in Ref. 8, but is assumed to be approximately 288°C (550°F).
2. Uniform elongation values were estimated from the SSRT stress-strain curves in Ref. 8 (shown as Figure 3-6).
3. The stress-strain curve shows that this specimen either fractured before necking or the test was terminated before necking. This appears to be the reason why its uniform elongation and total elongation values are significantly below the typical values for this fluence level.

Halden BHWR Reactor - NUREG/CR-6892[9]

The 2006 NUREG/CR-6892 report listed SSRT results of many heats of solution annealed and water quenched Type 304 and Type 316 materials irradiated in the Halden boiling heavy water reactor (BHWR) in Norway. The SSRT specimens were machined before being irradiated at 288°C (550°F) in Halden. The SSRT specimen geometry is shown in Figure 3-7. Table 3-5 summarizes the SSRT results tested in air of a commercial purity Type 304SA irradiated to moderate fluence.

Table 3-5 SSRT Results of Commercial Purity Type 304SA Irradiated in Halden BHWR[9]

Irradiation Temperature 288°C (550°F), Tested in Air, Strain Rate 1.65 x 10^{-7}/sec.

Specimen ID	SSRT No.	Test Temp.		Fluence	0.2% Yield	Ultimate Tensile	Uniform Elong.	Total Elong.
		°C	°F	n/cm² E > 1 MeV	MPa	MPa	%	%
C19-05	HR-58	23	73	9 x 10^{20}	1130	1260	15.50	15.70
C19-02	HR-30	289	552	9 x 10^{20}	888	894	6.41	10.21

Advanced Test Reactor – Jacobs et al.[10]

The 1987 paper by Jacobs et al. listed constant elongation rate tensile (CERT) test results of several types of austenitic stainless steels irradiated in the Advanced Test Reactor (ATR) in Idaho. The ATR is a pressurized, light-water moderated and cooled with a beryllium reflector. The CERT specimens were irradiated at 500-625°F (260-329°C) in ATR. The CERT specimen geometry is shown in Figure 3-8. Table 3-6 summarizes the CERT results tested in air at 288°C (550°F) of a solution annealed and water quenched Type 304.

Table 3-6 CERT Results Type 304SA Irradiated in ATR [10]

Irradiation temperature 500-625°F (260-329°C), CERT in air at strain rate 4 x 10^{-4}/sec.

	Test Temp.		Fluence	0.2% Yield	Ultimate Tensile	Uniform Elong.	Total Elong.
	°C	°F	n/cm² E > 1 MeV	ksi	ksi	%	%
304, Solution Annealed	288	550	0	24.9	64.3	41.3	48.0
304, Solution Annealed	288	550	8 x 10^{20}	80.2	92.7	12.6	17.5

JMTR and Halden BHWR – Navas et al.[11]

The 2003 paper by Navas et al. provided SSRT stress-strain curves for two heats of Type 304SA. One heat contained 0.05% carbon (HC) and the other contained 0.012% carbon (ULC). Both heats received a sensitization heat treatment prior to irradiation. The sensitization treatment was 100 minutes at 750°C (1382°F) followed by 24 hours at 500°C (932°F). After this treatment, the HC heat had a continuous chromium carbide precipitation along grain boundaries while the ULC heat had few and discontinuous carbide precipitates at grain boundaries. The ASTM A 262 Practice A screening test for sensitization showed a "ditch" microstructure for the HC heat and a "dual" microstructure for the ULC heat. The SSRT specimen geometry is shown Figure 3-9.

Neutron irradiation of the SSRT specimens was performed in the Japan Materials Testing Reactor (JMTR) and Halden reactor in Norway. The JMTR is a light-water type test reactor used for testing materials for BWRs and PWRs. The SSRT specimens were irradiated at 290°C (554°F) to 7.0x10^{19} and 1.1x10^{20} n/cm², E>1 MeV in the JMTR reactor and to 5 x10^{20} and 1.2 x10^{21} n/cm², E>1 MeV in the Halden reactor. The typical SSRT stress-strain curves in inert gas at 290°C (554°F) are shown in Figure 3-10. The SSRT strain rate was 3.5x10^{-7}/sec. The uniform and total elongations estimated from the Figure 3-10 SSRT stress-strain curves are listed in Table 3-7.

Table 3-7 Uniform and Total Elongations of Type 304SA Irradiated in JMTR and Halden Reactors [11]

Irradiation temperature 290°C (554°F), SSRT in inert gas at strain rate 3.5 x 10^{-7}/sec.

Heats	Test Temp.		Fluence	Uniform Elong.	Total Elong.
	°C	°F	n/cm² E > 1 MeV	%	%
Type 304 HC (0.05%C)	290	554	0	48	51
			7 x 10^{19}	34	37
			1.1 x 10^{20}	38	40
			5 x 10^{20}	22	25
			1.2 x 10^{21}	15	18
Type 304 ULC (0.012%C)	290	554	0	48	54
			7 x 10^{19}	40	43
			1.1 x 10^{20}	36	42
			5 x 10^{20}	27	37

3.4 *Evaluation*

Effect of Strain Rate

A major source of the irradiated tensile data reviewed in Section 3.3 is from SSRT or CERT testing. When tested in simulated BWR and PWR water, SSRT (CERT) tests are used for evaluating irradiation-assisted stress corrosion cracking (IASCC) of stainless steels in BWR and PWR reactor internals. The Type 304SA data in Section 3.3 are from the baseline SSRT specimens tested in air at elevated temperature. The effect of strain rate on the tensile properties of irradiated Type 304SA is shown in Figure 3-11 for room temperature, 450°F, and 700°F.[12] Uniform elongation is seen to decrease moderately with decreasing strain rate at elevated temperatures of 450°F and 700°F. Hence, the uniform elongation values from SSRT at ~10^{-7}/sec are conservative compared to those obtained from conventional tensile tests at a strain rate typically ranging from ~10^{-4} to ~10^{-2}/sec.

Comparison of BAW-10008 Lines with Recent Test Data

Figure 3-12 compares the uniform elongation lines in Figure 2-1 (Figure E-3 of Appendix E) with more recent data reviewed in Section 3.3. The recent Type 304SA data are from a decommissioned PWR, several BWRs, a heavy water reactor test reactor (Halden), a light water test reactor (JMTR), and a pressurized light water test reactor (ATR). The neutron flux levels of these reactors are more representative of PWR internals than fast breeder reactors. Therefore, the reported fluence and the associated tensile data from these reactors are more representative of PWR internals than those from fast breeder reactors used for establishing the lines in Figure E-3 of Appendix E.

The 300°C and 400°C lines in Figure 2-1 form the lower bound for the recent test data between fluence 10^{20} and 10^{21} n/cm^2, E>1 MeV. Therefore, the BAW-10008 lines have been confirmed to be conservative for the bottom core barrel flange. After neutron exposure of [ ], the anticipated minimum uniform elongation of Type 304SA is [ ] per the 300°C line and [ ] per the 400°C line. Therefore, the minimum uniform elongation of the bottom core barrel flange at the end of a 60-year lifetime is predicted to be [ ] at its operating temperature of 560°F (293°C). The typical uniform elongation of the bottom core barrel flange at the end of a 60-year lifetime is predicted to be [ ] at operating temperature based on the recent test data shown in Figure 3-12.

Deformation Limits in BAW-10008

The "deformation limits" in Appendix E were examined in Appendix A to BAW-10008, which provided the bases for the allowable stress limits for the BAW-10008 analysis. The last two paragraphs of Appendix A contain the following discussion on plastic strain limit:

> *"Stresses compared to the 2/3 S_u allowable are calculated on an elastic basis. If a plastic analysis of an internals component is performed, the plastic strains are limited to the strain corresponding to 2/3 S_u.*
>
> *Figure A-2 shows a minimum stress-strain curve for 304 stainless steel at 600°F. The strain corresponding to a stress of 2/3 S_u is 8.6%, which is approximately equal to 0.20 x uniform strain, or 7.6%."*

Figure A-2 in BAW-10008, shown as Figure 3-13 in this document, shows the minimum stress-strain curve of unirradiated Type 304SA at 600°F. According to Appendix A, a plastic strain would be limited to the strain corresponding to 2/3 of the ultimate tensile strength (UTS). For unirradiated Type 304SA, this limit would be 7.6% plastic strain at 600°F for a plastic analysis. The 2/3 UTS is approximately 42 ksi (290 MPa) at 600°F in Figure 3-13. Due to the irradiation hardening effect, the uniform elongation limit for irradiated Type 304SA would be lower than 7.6%. For example, the stress-strain curves in Figure 3-10 indicate that the plastic strain corresponding to 42 ksi (290 MPa) would be 3% or less for Type 304SA irradiated to above 10^{20} n/cm^2, E>1 MeV. In fact, as the fluence increases, the uniform plastic strain requirement is self-limiting, i.e., the plastic strain to 2/3 UTS will always be within the uniform elongation range. This is consistent with the original statement in Appendix E that "*As irradiation increases the yield strength to this value (i.e., 2/3 UTS), the plastic strains approach zero, ...*"

<u>Update of BAW-10008 Appendix E</u>

The bottom core barrel flange is predicted to have a minimum [ ] uniform elongation at operating temperature at the end of a 60-year lifetime. The typical uniform elongation at the end of a 60-year lifetime is predicted to be [ ] at operating temperature. Therefore, the original conclusion in BAW-10008 concerning the ductility for a 40-year lifetime remains valid for a 60-year lifetime. Hence, the last paragraph in Appendix E to BAW-10008 is updated to the following for a 60-year lifetime (the underline words have been updated):

> *The maximum fluence in the core barrel in the region near flanges <u>is conservatively estimated to be [ ] at the end of a 60-year (54 EFPY) lifetime.</u> This is the region of maximum stress intensity and the region where a loss of ductility would be detrimental. As noted in <u>Figure 3-12, the uniform elongation is predicted to be [ ] minimum and [ ] typical</u> for this fluence. It is concluded that even at the end of a <u>60-year lifetime</u>, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and the irradiation will not adversely affect deformation limits.*

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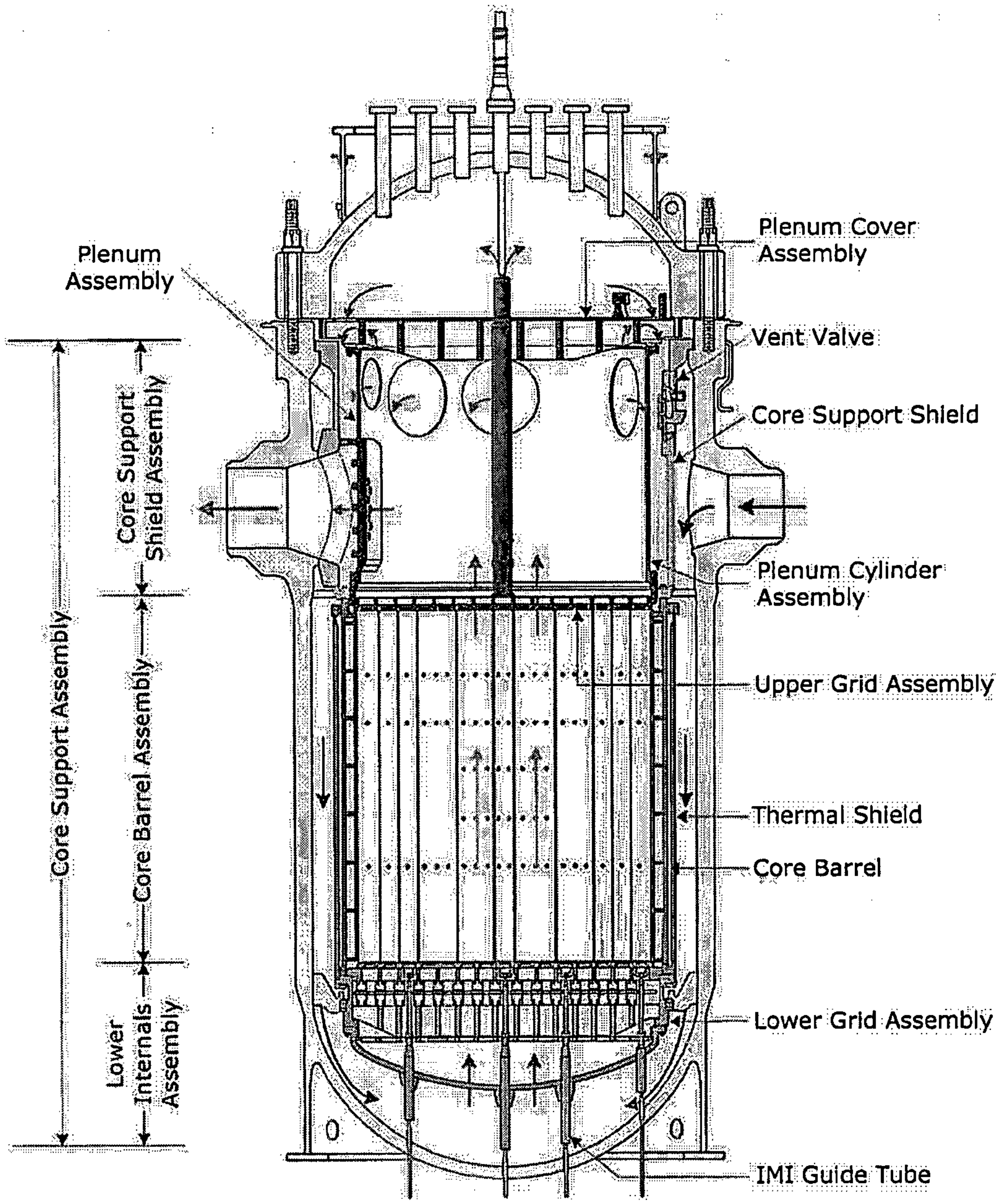


Figure 3-1 B&W-Designed PWR Internals General Arrangement (Note: some component items are rotated for clarity)

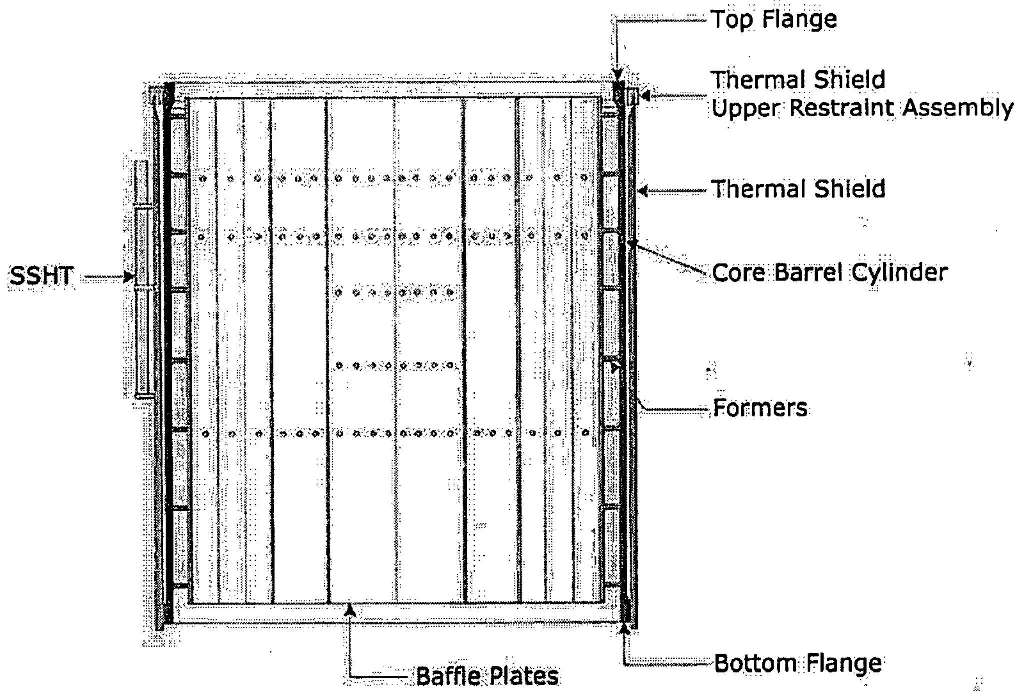


Figure 3-2 Locations of the top and bottom core barrel flanges in the B&W FA-177 core barrel assembly.

Figure 3-3 Tensile specimen machined from Type 304SA removed from a decommissioned PWR; all dimension unit in "inches".[6]

Figure 3-4 Stress-strain curve of Type 304SA core barrel [ ].[6]

[ ]

Figure 3-5 Tensile specimen machined from Type 304SA riser pipe removed from the [ ] BWR; all dimension unit in "mm".[7]

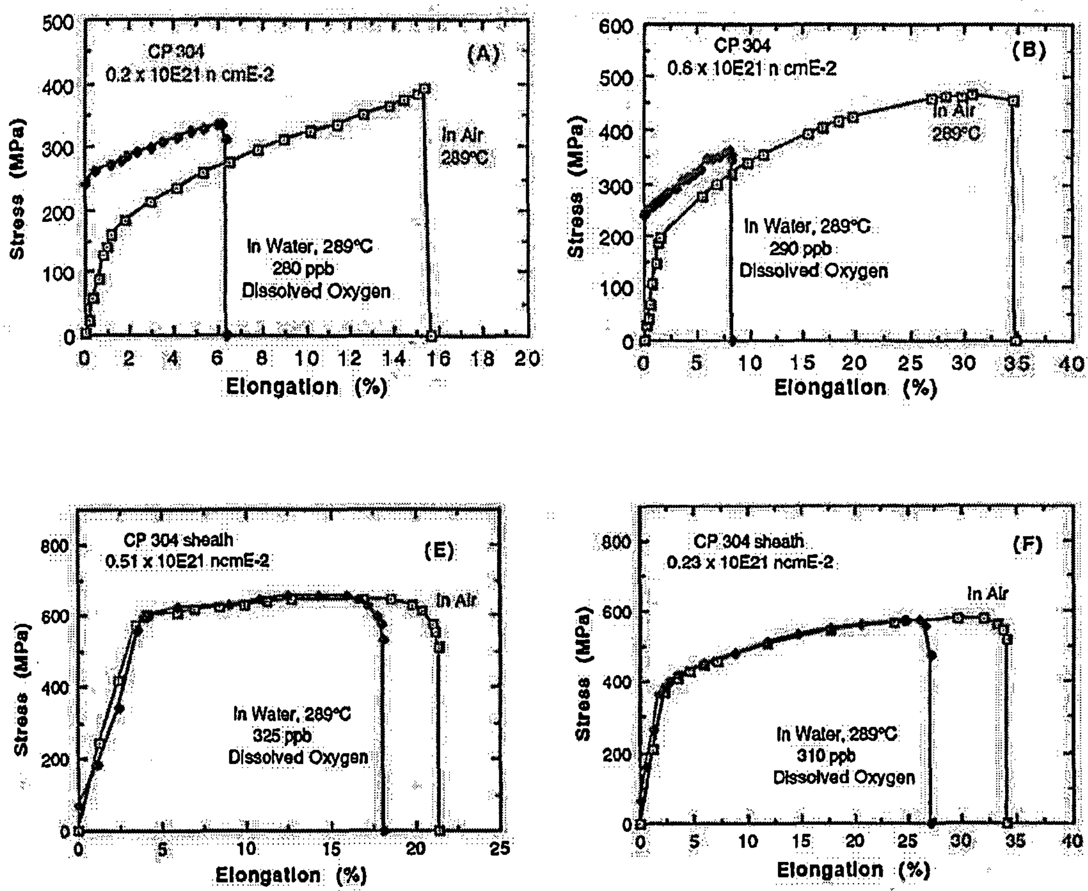


Figure 3-6 SSRT stress-strain curves of commercial purity Type 304SA removed from BWRs. Top, heat CP304-A BWR absorber tubes; bottom, heat CP304-B BWR control blade sheath. Strain rate 1.65 x 10^{-7}/sec.[8]

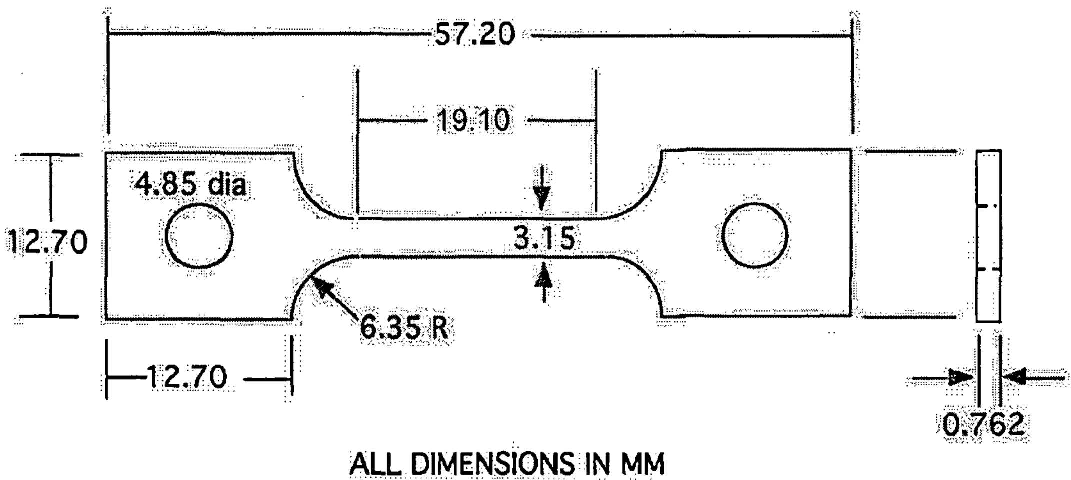


Figure 3-7 SSRT specimens irradiated in the Halden boiling heavy water test reactor.[9]

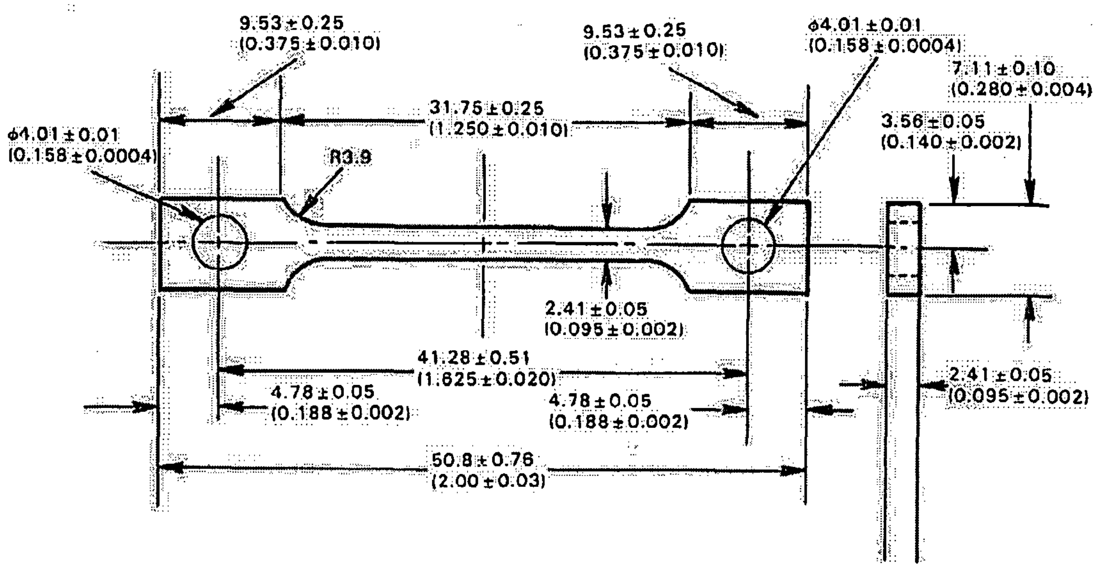
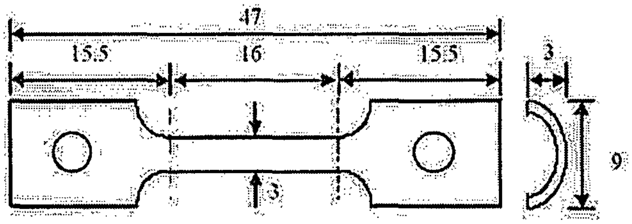
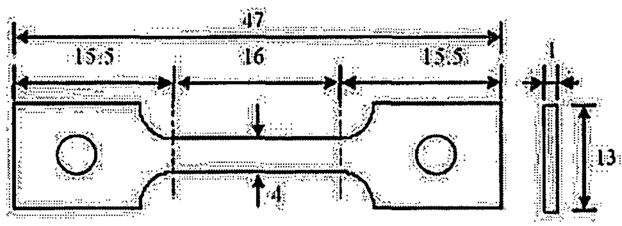


Figure 3-8 CERT specimens irradiated in the ATR pressurized light water test reactor. Dimensions are shown as mm (in.).[10]



a) Tube specimens of HC material



b) Plate specimens of ULC material

Figure 3-9 SSRT specimens irradiated in the JMTR light water test reactor and Halden BHWR reactor; all dimension in "mm".[11]

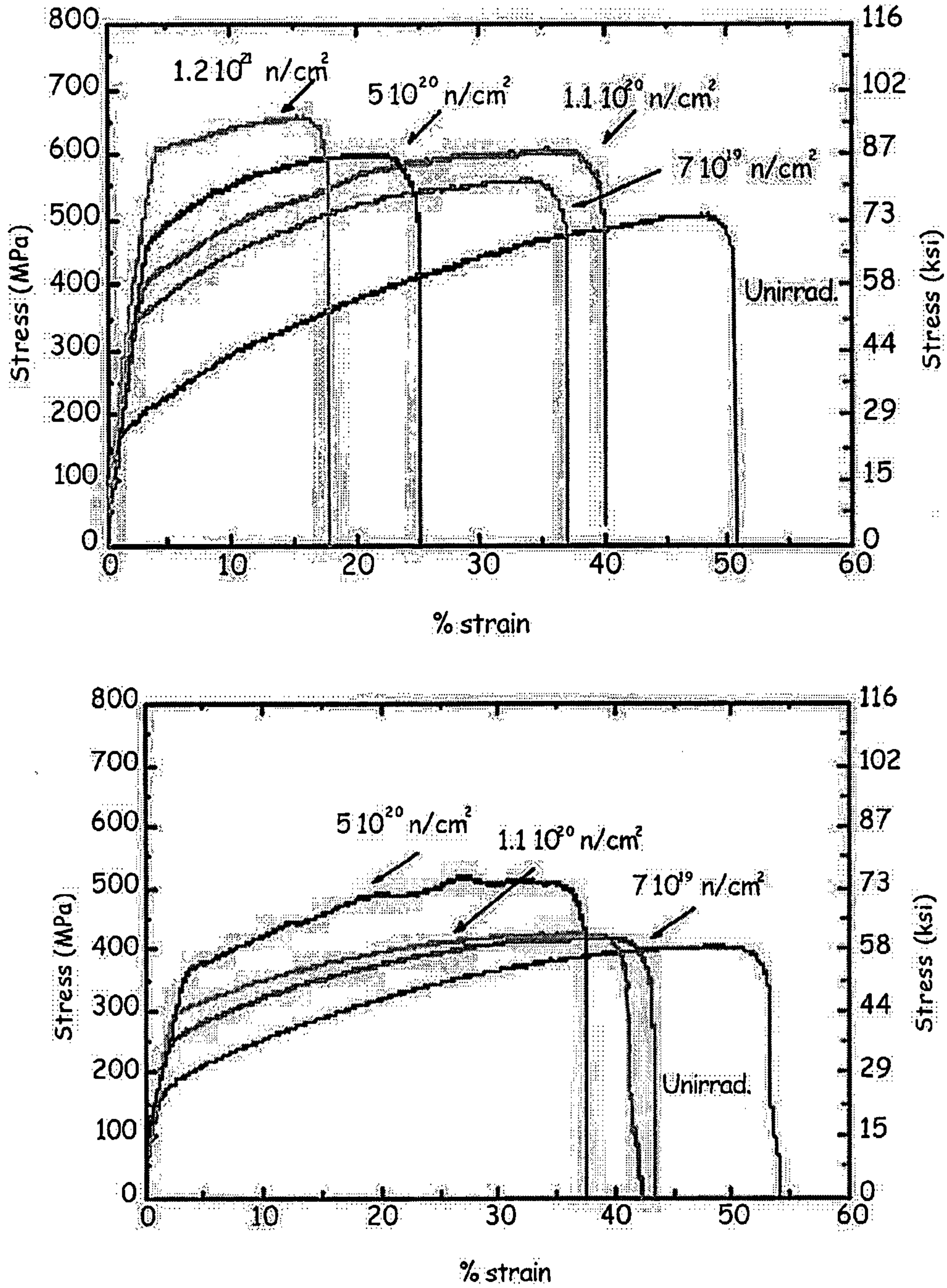


Figure 3-10 SSRT stress-strain curves of Type 304SA received a sensitization heat treatment prior to irradiation. Top Type 304 with 0.05% carbon heat, bottom Type 304 with 0.012% carbon. Irradiated in JMTR and Halden reactors at 290°C (554°F), tested in inert gas at 290°C (554°F), strain rate 3.5×10^{-7}/sec.[11]

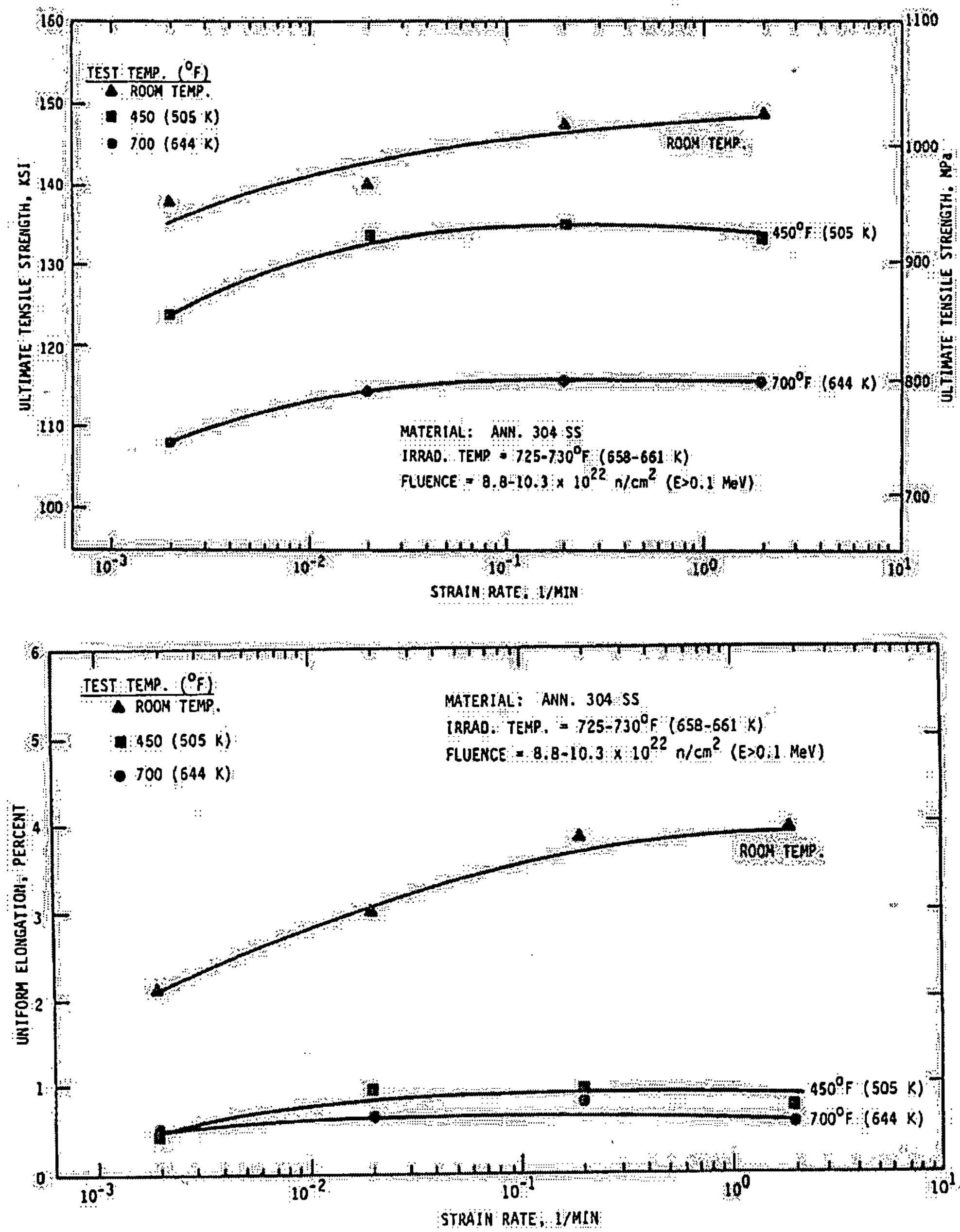


Figure 3-11 Effect of strain rate on tensile properties of Type 304SA. The material was irradiated at 725–730°F in EBR-II to 8.8–10.3 x 10^{22} n/cm^2, E > 0.1 MeV.[12]

(   )

Figure 3-12 Comparison of BAW-10008 uniform elongation lines with more recent test data of Type 304SA irradiated between 10^{20} and 10^{21} n/cm^2, E>1 MeV from water moderated reactors.

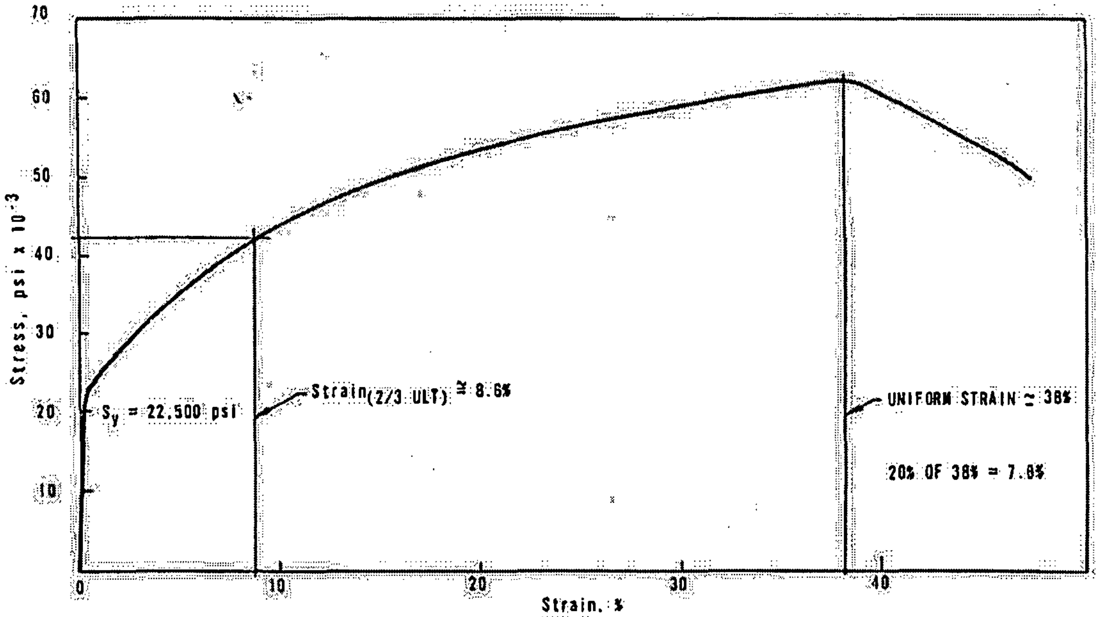


Figure 3-13, Minimum stress-strain curve of unirradiated Type 304SA stainless steel 600°C. (Figure A-2 of Appendix E of BAW 10008 Part 1, Rev. 1[3]).

4 Summary

This document has updated the estimated fluence of the core barrel flanges in BAW-10008, Appendix E to a 60-year lifetime (54 EFPY). The maximum 60-year fluence is [ ] for the top core barrel flange and [ ] for the bottom core barrel flange. The uniform elongation lines between fluence [ ] in Appendix E to BAW-10008 have been verified to be conservative with the recent Type 304SA data. These data are from a decommissioned PWR, several BWRs, a heavy water reactor test reactor (Halden), a light water test reactor (JMTR), and a pressurized light water test reactor (ATR). The neutron flux levels of these reactors are more representative of PWR internals than fast breeder reactors. Therefore, the reported fluence and the associated tensile data from these reactors are more representative of PWR internals than the data from fast breeder reactors used for establishing the lines in Figure E-3 of Appendix E.

The bottom core barrel flange is predicted to have a minimum [ ] uniform elongation at operating temperature at the end of a 60-year lifetime (i.e., the end of the license renewal period). The typical uniform elongation at the end of a 60-year lifetime is predicted to be [ ] at operating temperature. Therefore, the original conclusion in BAW-10008 concerning the ductility for a 40-year lifetime remains valid for a 60-year lifetime. Hence, the last paragraph in Appendix E to BAW-10008 is updated to the following for a 60-year lifetime (the underline words have been updated):

> *The maximum fluence in the core barrel in the region near flanges is conservatively estimated to be [ ] at the end of a 60-year (54 EFPY) lifetime. This is the region of maximum stress intensity and the region where a loss of ductility would be detrimental. As noted in Figure 3-12, the uniform elongation is predicted to be [ ] minimum and [ ] typical for this fluence. It is concluded that even at the end of a 60-year lifetime, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and the irradiation will not adversely affect deformation limits.*

5 References

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