Enclosure B

Letter L-17-026

Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse)

**<u>AREVA Report ANP-3542NP,
"Time-Limited Aging Analysis (TLAA) Regarding
Reactor Vessel Internals Loss of Ductility for
Davis-Besse Nuclear Power Station, Unit No. 1 at 60 Years"</u>**

<u>(Non Proprietary)</u>

31 pages follow

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AREVA

Time-Limited Aging Analysis (TLAA) Regarding Reactor Vessel Internals Loss of Ductility for Davis-Besse Nuclear Power Station, Unit No. 1 at 60 Years

ANP-3542NP
Revision 0

Licensing Report

December 2016

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

Contents

List of Tables

List of Figures

Nomenclature

Acronym	**Definition**
10 CFR	Title 10, Code of Federal Regulations
AMP	Aging Management Program
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
B&W	Babcock & Wilcox
DB-1 [a]	Davis-Besse
EFPY	Effective Full Power Year
FENOC	FirstEnergy Nuclear Operating Company
LBB	Leak Before Break
LOCA	Loss of Coolant Accident
LRA	License Renewal Application
MeV	Mega (Million) electron Volt
NRC	U.S. Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RV [a]	Reactor Vessel
RVI [b]	Reactor Vessel Internals
SER	Safety Evaluation Report
Sy	Yield Stress
TLAA	Time-Limited Aging Analysis
UCN	UFSAR Change Notice
UFSAR	Updated Final Safety Analysis Report

[a] Used in the direct quotes by FENOC.
[b] Used in the direct quotes by NRC.

ABSTRACT

This document provides the Davis-Besse Nuclear Power Station, Unit No. 1 plant-specific evaluation regarding the effect of irradiation on the mechanical properties and deformation limits of reactor vessel internals component items at the expiration of the renewed license, which is 60 years or 52 effective full-power years, as required per the amended License Renewal Application and Nuclear Regulatory Commission supplemental Safety Evaluation Report.

1.0 INTRODUCTION

FirstEnergy Nuclear Operating Company (FENOC) submitted the License Renewal Application (LRA) for Davis-Besse Nuclear Power Station, Unit No. 1 (Davis-Besse) in August 2010, which provided the technical information as required by Title 10, Part 54, of the Code of Federal Regulations (10 CFR) [(1)]. The application intended to provide sufficient information for the U.S. Nuclear Regulatory Commission (NRC) to complete its technical reviews for renewal of the Davis-Besse operating license for an additional 20 years. Section 4.2.7 of the Davis-Besse LRA describes the time-limited aging analysis (TLAA) for the reduction in fracture toughness of the reactor vessel internals and states that this TLAA will be managed during the period of extended operation through the Pressurized Water Reactor (PWR) Reactor Vessel Internals Program.

By letter dated April 21, 2015, FENOC amended the LRA and submitted the inspection plan for the reactor vessel internals components at Davis-Besse in order to fulfill the conditions and criteria specified in LRA Commitment No. 15, which was included as a commitment in Updated Final Safety Analysis Report (UFSAR) Supplement Table A-1 [(2)]. In the Davis-Besse Reactor Vessel Internals Inspection Plan [(2)], it was confirmed that the reduction of ductility fracture toughness analysis for the reactor vessel internals components is a TLAA for the LRA.

Based on a telephone conference call held with the NRC staff on May 6, 2015, to discuss the Davis-Besse Reactor Vessel Internals Inspection Plan details, FENOC submitted, by letter, supplemental information to support completion of the NRC review of the Plan [(3)]. In the letter, FENOC amended the Davis-Besse UFSAR Supplement Table A-1 to include Commitment No. 54, which was linked to the UFSAR Supplement summary description for the PWR Reactor Vessel Internals Program (i.e., LRA Section A.1.32), and in which the FENOC committed to submit an updated reduction of ductility fracture toughness analysis to the NRC for review and approval. The Commitment No. 54 action within the FENOC letter stated:

In response to MRP-227-A Applicant/Licensee Action Item 8, update and submit for NRC review and approval an evaluation for the period of extended operation regarding the effect of irradiation on the mechanical properties and deformation limits of the RV {reactor vessel} internals that was evaluated for the current term of operation in Appendix E of Topical Report BAW-10008, Part 1, Revision 1 supplemented by DB-1 {Davis-Besse} U{F}SAR Appendix 4A.

The NRC released a supplemental safety evaluation report (SER) for the Davis-Besse LRA [4] stating the following in Section 4.2.7.2:

Based on the above evaluation {SER Section 3.0.3.3.6 as supplemented by Section 3.0.3.3.6 of the Supplemental SER}, the staff finds that the applicant demonstrated, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of aging on the fracture toughness of the RVI {reactor vessel internals} will be adequately managed by the PWR Reactor Vessel Internals Program for the period of extended operation.

Additionally, the NRC's supplemental SER for the Davis-Besse LRA [4] concludes the following in Section 4.2.7.3:

On the basis of its review, the staff concludes that the applicant provided an acceptable demonstration, pursuant to 10 CFR 54.21(c)(1)(iii), that the effects of reduction in fracture toughness on the integrity of RVI {reactor vessel internals} components will be adequately managed for the period of extended operation. The staff also concludes that the U{F}SAR supplement contains an appropriate summary description of the TLAA evaluation, as required by 10 CFR 54.21(d), and, therefore, is acceptable.

As noted in UFSAR Change Notice (UCN) 16-145 [11], Davis-Besse License Renewal Commitment No. 54 has been updated to read as:

In response to MRP-227-A Applicant/Licensee Action Item 8, develop a schedule, with completion prior to the period of extended operation, for the update and submittal for NRC review and approval of an evaluation for the period of extended operation regarding the effect of irradiation on the mechanical properties and deformation limits of the RV internals that was evaluated for the current term of operation in Appendix E of Topical Report BAW-10008, Part 1, Revision 1 supplemented by DB-1 {Davis-Besse} U{F}SAR Appendix 4A.

This document provides the plant-specific evaluation of ductility for the Davis-Besse reactor vessel internals at the expiration of the renewed license (i.e., 60 years or 52 effective full power years (EFPYs)) as required by the amended LRA and NRC supplemental SER.

2.0 BACKGROUND

A plant-specific TLAA regarding reduction of fracture toughness of (stainless steel) reactor vessel internals is addressed in the Davis-Besse reactor vessel internals inspection plan prepared by AREVA, Inc. for FENOC [(2)]. Section 4.2.4.1 of this inspection plan addresses currently-identified TLAAs, including reduction of fracture toughness of the reactor vessel internals. The LRA for Davis-Besse [(1)], Section 4.2.7, states that this TLAA is managed by the PWR Reactor Vessel Internals Program for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). The program (i.e., MRP-227-A) is now approved and is implemented via the Davis-Besse inspection plan as part of the Davis-Besse reactor vessel internals aging management program (AMP).

The Davis-Besse UFSAR, Section 4.2.2, references the topical report BAW-10008 Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," [(5)] as supplemented by Davis-Besse UFSAR Appendix 4A, "Faulted Condition Analysis of Toledo, Davis-Besse Reactor Internals." Davis-Besse UFSAR, Appendix 4A, describes the detailed stress analysis specific to the Davis-Besse reactor vessel internals for faulted conditions since BAW-10008 Part 1, Revision 1 is only applicable to reactor vessel lowered loop, skirt supported Babcock & Wilcox (B&W) unit designs, while Davis-Besse is a raised loop, nozzle supported B&W unit. The Davis-Besse specific reactor vessel internals stress intensity values for normal, upset, and faulted loads are reported in UFSAR Section 4.2.2, Table 4.2-5.

The effect of irradiation on the mechanical properties and deformation limits for the reactor vessel internals was evaluated for 40 years in Appendix E of topical report BAW-10008 Part 1, Revision 1, for the lowered loop, skirt supported B&W units. The evaluation in Appendix E concluded that the reactor vessel 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. Specifically, the maximum fluence in the core barrel in the region near flanges is much less than 10^{20} n/cm^2, E > 1 mega (million) electron volt (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.

A specific assessment of the effect of irradiation on mechanical properties and deformation limits for the Davis-Besse reactor vessel internals relative to Appendix E of BAW-10008, Part 1, Revision 1 is not included in the Davis-Besse UFSAR, Appendix 4A. However, FENOC concluded in their LRA that the TLAA reported in BAW-10008 Part 1, Revision 1, Appendix E, is applicable to Davis-Besse owing to the load and stress intensity reconciliation to BAW-10008 Part 1, Revision 1, reported in Sections 4.2.2 and Appendix 4A of the Davis-Besse UFSAR.

3.0 ASSUMPTIONS

There are no assumptions for this document.

4.0 INPUTS

The Davis-Besse core support components are designed to meet the stress requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, in effect on the date of order for normal operation and transients. A detailed stress analysis of the Davis-Besse reactor vessel internals under accident conditions has been completed using the methods reported in Appendix 4A, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake" of the Davis-Besse UFSAR. The results of this analysis are shown in Table 4.2-5 of the Davis-Besse UFSAR and are reproduced in Table 4-2.

The unirradiated ASME B&PV Code yield stress values at 600°F for the Davis-Besse reactor vessel internals component items materials of interest, [ ], are presented in Table 4-1.

Table 4-1
ASME B&PV Code Yield Stress Values at 600°F

Temperature, °F	ASME Code Yield Stress (Sy), psi (Note 1)			
	Type 304/CF8 Stainless Steel	Type 304L Stainless Steel	Alloy A-286 (Note 2)	Alloy X-750 (Note 3)
600	[ ]	[ ]	[ ]	[ ]

Note 1: [

]

Note 2: [ ]

Note 3: [

]

Table 4-2
Stress Summary for Davis-Besse Reactor Internals (Davis-Besse UFSAR Table 4.2-5)

TABLE 4.2-5

Stress Summary for Davis-Besse Internals

Comments	Stress type	Case I: Normal Stress intensity, psi	Case I: Normal Allowable stress, psi	Case II: Upset Stress intensity, psi	Case II: Upset Allowable stress, psi	Case III: Faulted Stress intensity, psi	Case III: Faulted Allowable stress, psi
Lower grid plate							
Outlet pipe rupture	P_L+P_b	300	24,000	450	28,800	24,200	37,500
Inlet pipe rupture	P_L+P_b	300	24,000	450	28,800	15,650	37,500
Plenus cover	P_L+P_b	1,300	24,500	2,800	28,000	36,550	37,500
Plenus cylinder reinforcing plate	P_L+P_b	900	23,400	1,000	28,000	14,400	37,500
Upper guide tubes	P_m	125	15,600	250	18,700	6,750	37,500
	P_L+P_b	405	23,400	810	28,000	11,910	37,500
Upper guide tube sectors	P_L+P_b	50	23,400	100	28,000	4,700	37,500
Core support shield top flange							
Subcooled part of LOCA	P_m	700	15,600	1,380	18,700	19,250	37,500
	P_L+P_b	950	23,400	1,400	28,000	30,650	37,500
Saturated part	P_m	700	15,600	1,380	18,700	12,050	37,500
	P_L+P_b	950	23,400	1,400	28,000	33,200	37,500
Core support shield lower flange							
Subcooled part of LOCA	P_m	1,010	15,600	2,125	18,700	16,000	37,500
	P_L+P_b	1,150	23,400	2,650	28,000	18,200	37,500
Saturated part	P_m	1,010	15,600	2,125	18,700	10,800	37,500
	P_L+P_b	1,150	23,400	2,650	28,000	18,650	37,500
Core barrel top flange							
Subcooled part of LOCA	P_m	800	15,600	1,700	18,700	16,000	37,500
	P_L+P_b	850	23,400	1,750	28,000	17,000	37,500
Saturated part	P_m	800	15,600	1,700	18,700	10,600	37,500
	P_L+P_b	850	23,400	1,750	28,000	23,000	37,500
Baffle plates	P_L+P_b	14,800	23,400	14,800	28,000	27,450	37,500
Internal bolts							
Core barrel-core	P_m	40,300	54,000	40,700	82,000	55,000	82,000
Support shield joint	P_L+P_b	46,100	81,000	46,500	82,000	77,900	82,000

NOTE: Qualification of the reactor vessel internal components with the integrated head assembly (IHA) installed is discussed in section 3.9.1.5.

4.2-43 UFSAR Rev 30 10/2014

5.0 ANALYSIS

Each reactor vessel internals component item listed in Table 4.2-5 of the Davis-Besse UFSAR is assessed in accordance with 1 of 4 categories to determine if each reactor vessel internals component item should be considered potentially susceptible to an unacceptable amount of loss of ductility at 60 years/52 EFPY. The 4 categories to determine the impact of loss of ductility at 60 year/52 EFPY on each item are as follows:

Category #1: The Case III faulted stress intensity listed in Table 4.2-5 of the Davis-Besse UFSAR for the reactor vessel internals component item is less than the unirradiated ASME B&PV Code yield stress at operating temperature (600°F), and therefore plasticity will not occur at 60 years/52 EFPY. Since the material remains elastic (and neutron embrittlement would increase the yield stress), loss of ductility is acceptable.

Category #2: The reactor vessel internals component item is already highly irradiated [ ], such that the Case III faulted stress intensity remains below the irradiated yield stress (increases as fluence increases) and plasticity will not occur at 60 years/52 EFPY. Since the material remains elastic, with large margin to the irradiated yield stress, the loss of ductility is acceptable.

Category #3: The neutron fluence is low and embrittlement is negligible, loss of ductility is minimal or will not occur at 60 years/52 EFPY and unirradiated ductility properties are still applicable.

Category #4: Potentially other justifications identified during the assessment (e.g., the component item, or embrittled portion thereof of the component item, is in compression during the faulted condition) such that loss of ductility is not applicable to the reactor vessel internals component item.

If the reactor vessel internals component item does not fall into any of the 1 through 4 categories above, further evaluation is needed.

5.1 *Assessment Per Criteria Defined by Category Item #1*

For the Category Item #1 assessment, the Case III faulted stress intensity listed in Table 4.2-5 of the Davis-Besse UFSAR for the reactor vessel internals component item is compared to the unirradiated ASME B&PV Code yield stress for the applicable material type at operating temperature (i.e., about 600°F). For those reactor vessel internals component items in Table 4.2-5 of the Davis-Besse UFSAR that have a reported Case III faulted stress intensity value less than the unirradiated ASME B&PV Code yield stress at 600°F, the material should remain elastic (and neutron embrittlement would increase the yield stress), loss of ductility is acceptable.

Of the 22 listed applicable reactor vessel internals component item stress types ("Pm" and/or "Pm + Pb") in Table 4.2-5 of the Davis-Besse UFSAR, 13 stress intensity values are less than the unirradiated ASME B&PV Code yield stress at 600°F, thus loss of ductility is acceptable for these component item stress types. For the remaining 9 reactor vessel internals component item stress types, further assessment in accordance with criteria defined by Category Item #2 was performed to determine if the applicable reactor vessel internals component item should be considered potentially susceptible to an unacceptable amount of loss of ductility at 60 years/52 EFPY.

5.2 *Assessment Per Criteria Defined by Category Item #2*

For the Category Item #2 assessment, the expected fluence (E > 1.0 MeV) exposure of the applicable reactor vessel internals component items are reviewed to determine if the component item is highly irradiated to the level where saturation of the material's yield stress has occurred as shown in Figure 5-1 (Figure 13(a) of NUREG/CR-7027 [(6)]).

[

] For the case where the component item is highly irradiated at 60 years/52 EFPY and the Case III faulted stress intensity is below the irradiated yield stress (which increases as fluence increases), plasticity will not occur at 60 years/52 EFPY; therefore the material remains elastic with large margin to the irradiated yield stress, and the loss of ductility is acceptable.

Based on the fluence estimates for the 9 stress types of the reactor vessel internals component items where the Case III faulted stress intensity values are greater than the unirradiated ASME B&PV Code yield stresses at 600°F, only the [ ] are expected to be highly irradiated [ ] If the [ ] estimated fluence is applied to the curve in Figure 5-1 (Figure 13(a) of NUREG/CR-7027), the irradiated yield stress is expected to be saturated at ~800 MPa (116,000 psi). Comparing the [ ] Case III faulted stress intensity in Table 4.2-5 of the Davis-Besse UFSAR [ ], the reported stress intensity is less than this irradiated saturated yield stress, therefore the material remains elastic with large margin to the irradiated yield stress, and the loss of ductility is acceptable.

‡ Calculated using the light water reactor (LWR) conversion factor of 10^{22} n/cm^2 (E > 1.0 MeV) = 15 dpa (NUREG/CR-7027).

For the remaining 8 stress types of the reactor vessel internals component items where the Case III faulted stress intensity values are greater than the unirradiated ASME B&PV Code yield stresses at 600°F, further assessment in accordance with criteria defined by Category Item #3 was performed to determine if the reactor vessel internals component item should be considered potentially susceptible to an unacceptable amount of loss of ductility at 60 years/52 EFPY.

5.3 *Assessment Per Criteria Defined by Category Item #3*

For the 8 remaining Davis-Besse reactor vessel internals component items stress types, a review was performed to determine if the expected fluence (E > 1.0 MeV) exposure of the reactor vessel internals component item is low enough such that neutron embrittlement is considered negligible, thus loss of ductility is minimal or will not occur at 60 years/52 EFPY for the component item.

If the projected fluences of the remaining reactor vessel internals component items are applied to Figure 5-2 (Figure E-3 of BAW-10008, Part 1, Revision 1), the decrease in uniform elongation for the [

] (all fabricated from Type 304 stainless steel) at both 572°F (300°C) and 752°F (400°C) (i.e., temperatures between which these component items would be expected to experience) is such that the 20 percent uniform elongation of irradiated material credited for 40 years in Appendix E of BAW-10008, Part 1, Revision 1 and the 8.6 percent allowable strain specified in Appendix A of BAW-10008, Part 1, Revision 1 is met for these component items.

Note that Figure 5-3 (Figure 3-12 in Reference (7)) provides recent irradiated Type 304 stainless steel solution annealed test data, which validates the conservatism of the curves in Figure E-3 of BAW-10008, Part 1, Revision 1. The slight decrease in uniform elongation at the projected fluence levels for the [ ] is confirmed in Figure 5-4 (Figure 13(c) of NUREG/CR-7027).

In addition, as shown in Figure 5 of Reference (8), the uniform elongation of unirradiated Type 304 solution annealed stainless steel at 600°F only decreases slightly with increasing strain rate. Further observation of the data within the figure shows that even at the highest tested strain rates (10^1/sec and 10^2/sec) at 600°F, the uniform elongation is above the 20 percent uniform elongation of irradiated austenitic stainless steel material credited for 40 years in Appendix E and the 8.6 percent allowable strain specified in Appendix A of BAW-10008, Part 1, Revision 1, which are considered applicable for Davis-Besse based on statements in Appendix 4A of Davis-Besse UFSAR regarding the limits in BAW-10008, Part 1, Revision 1. It is also observed that yield stress increases with increasing strain rate at 600°F as shown in Figure 3 of Reference (8). In addition to having sufficient ductility at 60 years/52 EFPY relative to the allowable stresses of BAW-10008, Part 1, Revision 1 and Table 4.2-5 of the Davis-Besse UFSAR, [ ] will have greater resistance to plastic deformation at increased strain rates.

For the stress types of the 2 remaining reactor vessel internals component items [ ] further assessment in accordance with criteria defined by Category Item #4 is performed to determine if the reactor vessel internals component item should be considered potentially susceptible to an unacceptable amount of loss of ductility at 60 years/52 EFPY.

5.4 *Assessment Per Criteria Defined by Category Item #4*

For the remaining 2 reactor vessel internals component items stress types [ ] no other justification is identified to credit the component item's loss of ductility as minimal or not applicable. Thus further assessment is needed to determine if the reactor vessel internals component item should be considered potentially susceptible to an unacceptable amount of loss of ductility at 60 years/52 EFPY.

5.5 *Further Assessment for [ ]*

Based on the results documented in Sections 5.1 through 5.4, further assessments are necessary for the [ ]

To further assess these 2 component items, a recalculated set of loads were developed to determine the faulted condition stresses. These recalculated loads were based on 2 separate methodologies: 1) stresses as a result of pipe break loadings developed due to the consideration of asymmetric effects (9) and 2) crediting leak-before-break (LBB) by eliminating primary loop pipe breaks from consideration (10); utilizing these recalculated loads, the faulted condition stress intensity may be reduced.

The faulted condition loads and stress intensity of the [ ] were recalculated crediting LBB of the primary loop piping, and the faulted condition loads and stress intensity of the [ ] were recalculated considering asymmetric pipe break loadings effects (i.e., developed after the data reported in Davis-Besse UFSAR, Appendix 4A, and UFSAR Table 4.2-5).

The unirradiated ASME B&PV Code yield stress value at 600°F for the reactor vessel internal component items material types of interest is the same as that used in Section 5.1 Category Item #1 assessment.

The recalculated faulted condition "Pm" and/or "Pm + Pb" stress intensity values crediting LBB for the [ ] and the more recent asymmetric LOCA loading stress calculation for the [ ] are less than the unirradiated ASME B&PV Code yield stress value at 600°F [ ] thus loss of ductility is acceptable for the [ ]

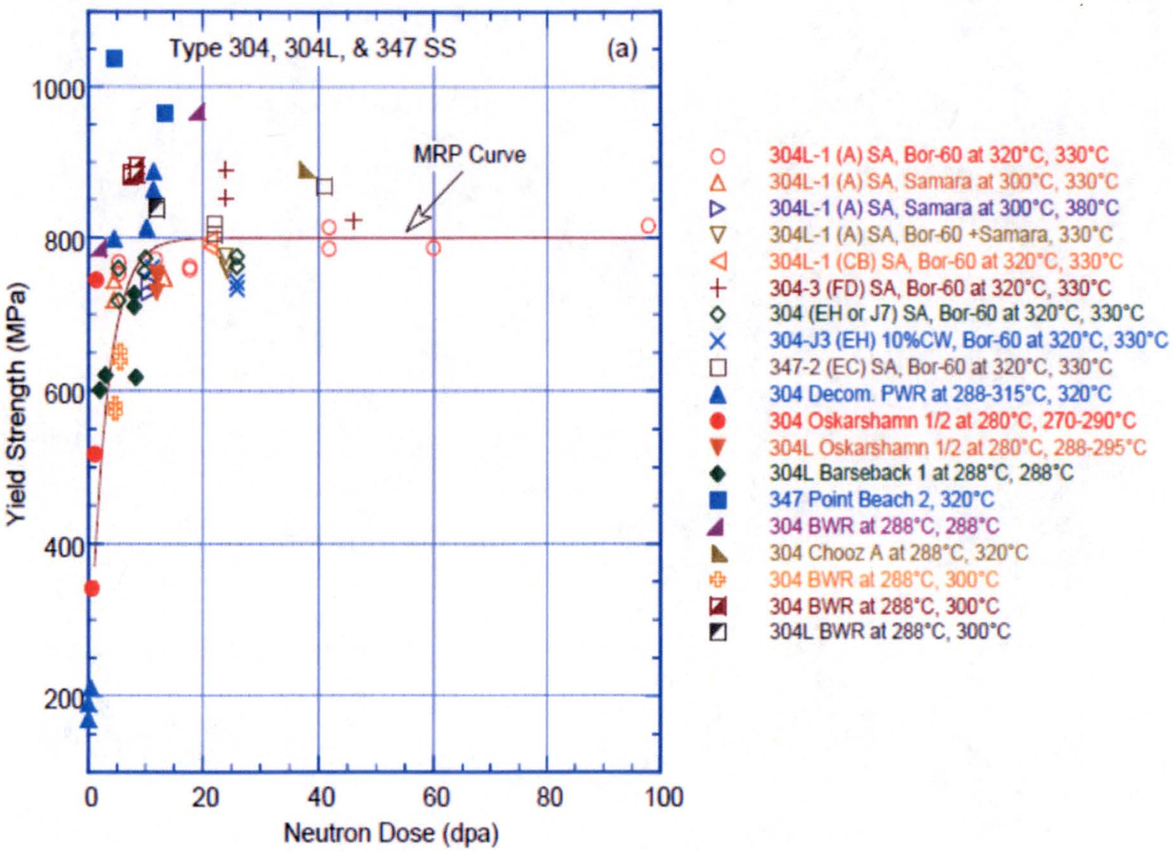

Figure 5-1
Change in Yield Strength as a Function of Neutron Dose for Solution-Annealed Type 304, 304L, and 347 Stainless Steels at Elevated Temperature, 270-380°C (Figure 13(a) of NUREG/CR-7027)

[ ]

Figure 5-2

[ ]

[ ]

[ ]

Figure 5-3

[ ]

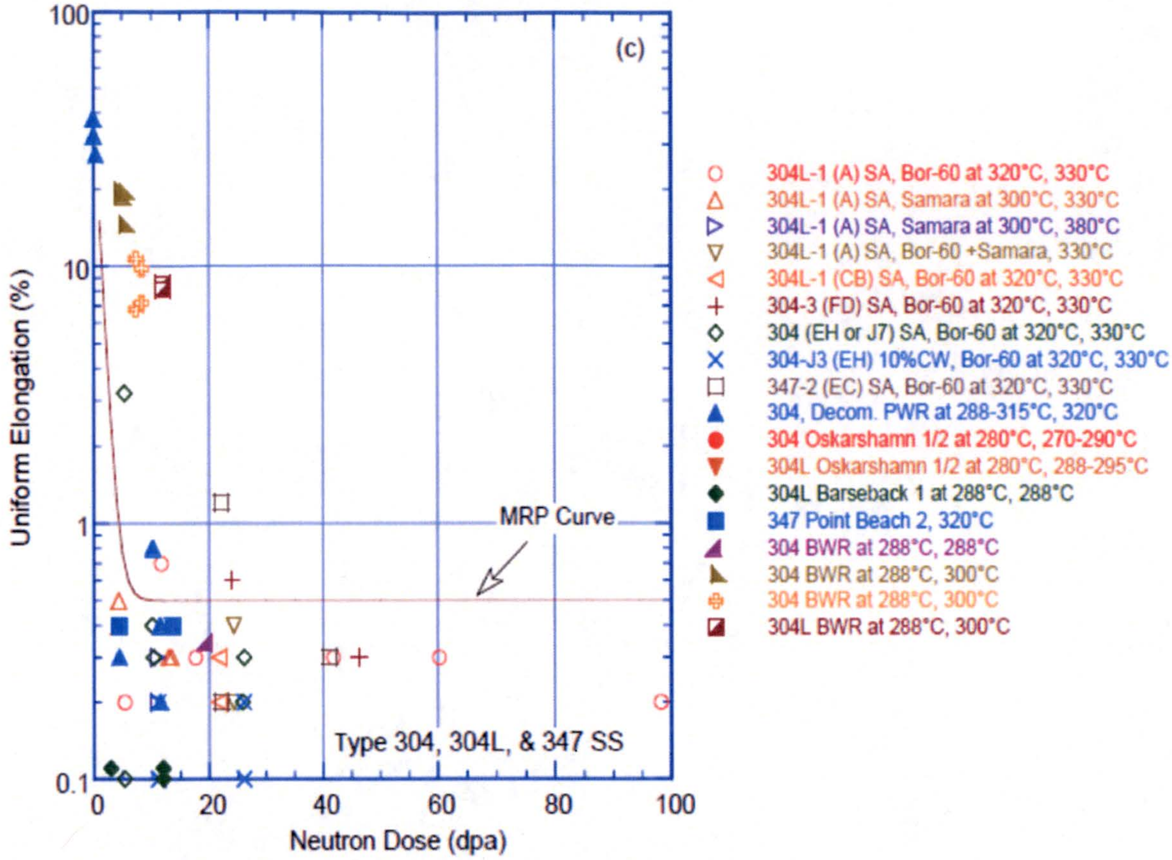

Figure 5-4
Change in Uniform Elongation as a Function of Neutron Dose for Solution-Annealed Type 304, 304L, and 347 Stainless Steels at Elevated Temperatures, 270-380°C (Figure 13(c) of NUREG/CR-7027)

6.0 RESULTS

The reactor vessel internals component items listed in Table 4.2-5 of the Davis-Besse UFSAR were assessed in accordance with 1 or more of 4 categories to determine if each component item should be considered potentially susceptible to an unacceptable amount of loss of ductility at 60 years/52 EFPY. Based on the results of the categorical assessments, 2 reactor vessel internals component items from Table 4.2-5 of the Davis-Besse UFSAR were determined to be potentially susceptible to an unacceptable amount of loss of ductility at 60 years/52 EFPY:

- [ ]
- [ ]

Follow-on assessments of these 2 reactor internals component items were performed, which included recalculated stress intensity values crediting LBB and asymmetric pipe break loading effects. These recalculated stress intensity values were then compared to the applicable unirradiated ASME B&PV Code yield stress at operating temperature, in accordance with the Category Item #1 methodology described in Section 5.1, to determine if they are still potentially susceptible to an unacceptable amount of loss of ductility at 60 years/52 EFPY. The comparison of the recalculated faulted condition stress intensity values to the applicable unirradiated ASME B&PV Code yield stress at 600°F shows the recalculated stress intensity values to be less than the unirradiated yield stress at 600°F for both component items.

Therefore, based on the assessments of the effect of irradiation on the mechanical properties and deformation limits of the Davis-Besse reactor vessel internals, the loss of ductility for the reactor vessel internals component items listed in Table 4.2-5 of the Davis-Besse UFSAR is considered acceptable and remains valid for a 60 year/52 EFPY lifetime for the Davis-Besse reactor vessel internals.

7.0 REFERENCES

1. FENOC Letter L-10-221 Transmitting Davis-Besse Nuclear Power Station, Unit No. 1, Docket No. 50-346, License Number NPF-3, License Renewal Application and Ohio Coastal Management Program Consistency Certification, dated August 27, 2010. (NRC Accession Nos. ML102450565, ML102450567, ML102450563, and ML102450568).
2. FENOC Letter L-15-139 Transmitting Davis-Besse Nuclear Power Station, Unit No. 1, Docket No. 50-346, License Number NPF-3, License Renewal Reactor Vessel Internals Inspection Plan (TAC No. ME4640), dated April 21, 2015. (NRC Accession Nos. ML15113B132, ML15113B133, and ML15113B134).
3. FENOC Letter L-15-166 Transmitting Davis-Besse Nuclear Power Station, Unit No. 1, Docket No. 50-346, License Number NPF-3, Supplemental Information for the Review of the Davis-Besse Nuclear Power Station, License Renewal Application Amendment No. 56, dated May 20, 2015. (NRC Accession No. ML15140A705).
4. NUREG-2193 Supplement 1, "Safety Evaluation Report Related to the License Renewal of Davis-Bess Nuclear Power Station, Docket Number 50-346," Date published: April 2016. (NRC Accession No. ML16104A350).
5. AREVA Document BAW-10008 Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," June 1970.
6. NUREG/CR-7027 (ANL-10/11), "Degradation of LWR Core Internal Materials due to Neutron Irradiation," December 2010. (NRC Accession No. ML102790482).

7. "Information in Support of the EPRI Materials Reliability Program (MRP): Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0) Review," October 29, 2010. (NRC Accession No. ML103090248).

8. "High Strain Rate Tensile Properties of AISI Type 304 Stainless Steel," J. M. Steichen, Journal of Engineering Materials and Technology, Volume 95, Issue 3, pp 182-185, July 1973.

9. NRC Letter from J.F. Stolz (NRC) to R.P. Crouse (TED), Asymmetric LOCA Loads - Safety Evaluation Report, 4-9-84 (Log Number 1488).

10. NRC Letter from J. F. Stolz (NRC) to J. Williams (TED), Safety Evaluation of B&W Owners Group Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, 2-18-86 (Log Number 1918). (NRC Accession No. 8603130124).

11. FENOC UFSAR Change Notice (UCN) 16-145, dated October 20, 2016.