ATTACHMENT 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION (RAI) RELATED TO THE MRP-227-A APPLICANT/LICENSEE ACTION ITEM #7 ANALYSIS SUPPORTING THE REACTOR VESSEL INTERNALS INSPECTION PLAN AMENDMENT REQUEST FOR OCONEE NUCLEAR STATION (ONS), UNITS 1, 2, AND 3

[NON-PROPRIETARY VERSION]

AREVA

Response to NRC Request for Additional Information (RAI) Related to the MRP-227-A Applicant/Licensee Action Item #7 Analysis Supporting the Reactor Vessel Internals Inspection Plan Amendment Request for Oconee Nuclear Station (ONS), Units 1, 2, and 3

ANP-3267Q1NP
Revision 0

March 2014
AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
Rev. 0	All	Original Issue

Contents

List of Tables

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Nomenclature

Acronym	Definition
CASS	Cast Austenitic Stainless Steel
DPA	Displacement Per Atom
EFPY	Effective Full Power Year
FIV	Flow-Induced Vibration
CRGT	Control Rod Guide Tube
IE	Irradiation Embrittlement
IMI	Incore Monitoring Instrumentation
LOCA	Loss of Coolant Accident
MRP	Materials Reliability Program
NRC	Nuclear Regulatory Commission
ONS	Oconee Nuclear Station
OD	Outer Diameter
PH	Precipitation-Hardened
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCS	Reactor Coolant System
RV	Reactor Vessel
SSE	Safe Shutdown Earthquake
TE	Thermal Aging Embrittlement

ABSTRACT

AREVA document ANP-3267, Revision 1, "MRP-227-A Applicant/Licensee Action Item #7 Analysis for the Oconee Nuclear Station Units," was prepared by AREVA for Duke Energy and subsequently submitted to the Nuclear Regulatory Commission (NRC) by Duke Energy. The NRC has issued Requests for Additional Information (RAIs) on this submittal, and this report provides the responses for RAIs 1 (all parts), 2 (all parts), 3 (all parts), 4, 5 (all parts), 6, 7, and 8.

1.0 INTRODUCTION AND SUMMARY

AREVA document ANP-3267, Revision 0, "MRP-227-A Applicant/Licensee Action Item #7 Analysis for the Oconee Nuclear Station Units," was prepared by AREVA for Duke Energy and subsequently submitted to the Nuclear Regulatory Commission (NRC) by Duke Energy (1). The NRC has issued Requests for Additional Information (RAIs) (2) on this submittal and this report provides the answers to those RAIs assigned to AREVA.

Upon receipt of the RAIs, Duke Energy and AREVA reviewed the RAIs and determined who would answer each RAI; the RAI responses were assigned to AREVA. Appropriate references pertaining to a specific RAI response are listed within the RAI response. Overall report references are listed in Section 3.0 of this report.

2.0 REQUESTS FOR ADDITIONAL INFORMATION (RAIs) AND RESPONSES

The NRC RAIs are reproduced from Reference (2) in Sections 2.1.1 through 2.8.1. The reference numbers cited in these sections are for the references listed in Reference (2). The AREVA responses are in Sections 2.1.2 through 2.8.2.

2.1 RAI-1

2.1.1 Statement of RAI-1

Section 3.2.3 of Report ANP-3267P, "MRP-227-A Applicant/Licensee Action Item #7 Analysis for the Oconee Nuclear Station Units," states, " [

] " Please confirm the following:

1. The loading used to determine "the amount of distortion allowed" has considered normal, faulted, and emergency conditions.
2. The "free path acceptance criterion" mentioned later in this section is based on this allowed distortion.
3. For completeness, the identifier and title of the report that documents this evaluation will be disclosed in the response.

2.1.2 Response to RAI-1

2.1.2.1 *Response to Part 1:*

In the finite element analysis, loading conditions used to determine operational displacement and distortion for the control rod guide tube (CRGT) consist of deadweight, preload, horizontal and vertical Loss of Coolant Accident (LOCA) and Safe Shutdown Earthquake (SSE) loads, and cross flow loading.

2.1.2.2 *Response to Part 2:*

[

]

2.1.2.3 *Response to Part 3*

AREVA would prefer not to provide this information in this response because the document is AREVA Proprietary. However, the NRC is welcome to view the document through appropriate means.

2.2 RAI-2

2.2.1 Statement of RAI-2

Section 3.3.2 states, "[

]" This appears to be contradictory to an earlier statement in this section that, "[

]" Usually, unacceptable excessive deformation is associated with excessive stresses. Please clarify the apparent contradiction by providing for the worst case spacer casting:

1. The loads on the spacer casting through the screws and the brazed connections and the resulting deformations before and after failure at one screw location, and any secondary loads on the spacer casting that were not mentioned in the report.
2. A description of the stress analysis methodology.
3. Discussion of the relative rigidity of the control rod guide tube (CRGT), screws, and the spacer casings to characterize the loads acting on the spacer castings and to explain the "[ ]" of the spacer casting mentioned in Section 3.4.
4. A sketch showing the CRGT, the screws, the spacer casting, and any parts of the brazement sub-assembly which are connected to the spacer casting to supplement your answer to Item (3).

2.2.2 Response to RAI-2

2.2.2.1 *Response to Part 1:*

The first statement above was a conclusion from the analysis. The second statement was hypothetical that preceded the analysis. [

]

[

]

Figure 2-1 and Figure 2-2 in this report show the deformation of an intact casting under the applied load and the deformation of a casting with one ligament failed including the removal of the screw load at that location.

2.2.2.2 *Response to Part 2:*

[

]

[ ]

2.2.2.3 *Response to Part 3:*

[

] The sketch provided in Figure 2-3 of this report illustrates the geometry.

[

] RAI-2 Part 4 shows sketch of casting and stress path lines. Table 2-1 of this report shows respective stresses for intact spacer casting #2 and one ligament failed spacer casting #1. [

]

2.2.2.4 *Response to Part 4:*

A sketch of the spacer casting configuration with stress path lines is provided in Figure 2-4 of this report.

Figure 2-1. Displacement Intact Spacer Casting

Figure 2-2. Displacement Casting With One Failed Ligament

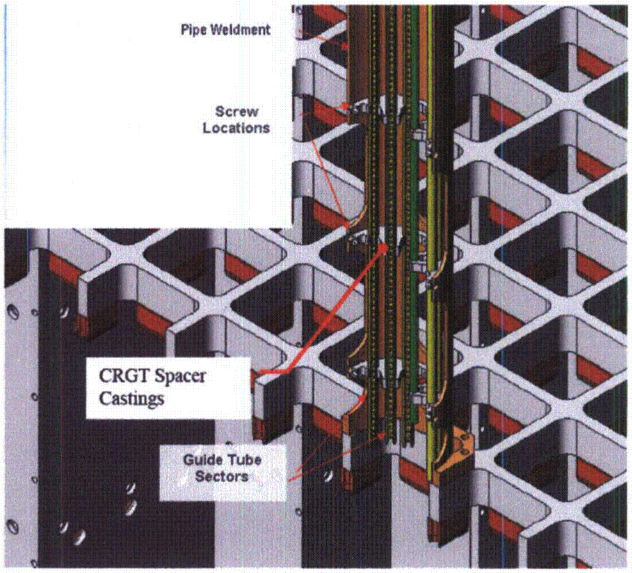

Figure 2-3. Brazement Subassembly Sketch

[

]

Figure 2-4. Spacer Casting Configuration with Stress Path Lines [

]

Table 2-1. Stress Distribution Around Castings

[ ]

2.3 RAI-3

2.3.1 Statement of RAI-3

Section 4.1.1 describes the incore monitoring instrumentation (IMI) guide tube spider castings. Please clarify the following:

1. This section states, "[ ]" Since original welding was performed on one leg after another and the weld residual load is a result from welding performed on other legs, please confirm that "[ ]" means "tend to partially unload."

2. This section also states, "[ ] " Please assess the severity of this unsatisfactory flow induced vibration and its consequence on the IMI guide tube system as a whole. Also, if all spider welds fail and the IMI guide tube is now carrying the spider mass, will the spider be pushed out the lower grid rib by flow and get stuck somewhere? Will the IMI guide tube still functional under this condition?

2.3.2 Response to RAI-3

2.3.2.1 *Response to Part 1:*

The exact weld sequence is not known but undoubtedly the spider was tack welded at least on each leg to the rib section pocket and then the welds were completed on the legs in some order. Regardless weld residual stresses are created by self-constraint of

the weld or inhibition of weld shrinkage of new welds due to the constraint of existing welds. It is agreed that removal of the weld on one side of the leg will tend to partially unload the weld on the opposite side and marginally decrease the load in the other legs.

[

]

2.3.2.2 *Response to Part 2:*

This statement is a carryover from the discussion of the original analysis of the incore guide tube. The guide tube is welded to the flow distributor head and then passes through a tapered forced fit arrangement and then finally has a slip fit with the Incore Monitoring Instrumentation (IMI) spider opening. The IMI spider itself is machined to a close fit with the lower rib section pocket and then all four legs are welded to the lower rib section pocket. [

] The incore instrument tube has a wear shield assembly that is all within the incore guide tube. The spider casting has a close slip fit with the outer diameter (OD) of the guide tube. [

]

2.4 RAI-4

2.4.1 Statement of RAI-4

Section 4.2.2 states, "[

]"

Explain how these percentages (i.e., [ ] % and [ ] %) were estimated and confirm that the same estimation methodology was applied to generate the percentages of saturation mentioned in the rest of Section 4.2.2.

2.4.2 Response to RAI-4

Best-estimate and bounding fluence value profiles (iso-fluence lines) for radial locations and vertical elevations within the RV internals were previously developed for RV bolting analyses on a generic basis. These profiles were used to determine the fluence values for areas of interest within the span of the IMI guide tube spider casting locations. The fluence values for a matrix of [

] were prepared to aid in the determination of expected fracture toughness properties (saturation) of the IMI guide tube spider castings at 32 Effective Full Power Years (EFPY), 48 EFPY, and 54 EFPY.

A recent evaluation of irradiation embrittlement (IE) of cast austenitic stainless steel (CASS) materials performed in MRP-276 does not identify any known CASS material fracture toughness tests results available for specimens irradiated at flux levels typical of Pressurized Water Reactors (PWRs), but limited test results for CASS specimens irradiated at high flux (i.e., in a fast breeder reactor) are available in NUREG/CR-7027 (Reference 1) (originally from Reference 2) and MRP-79, Revision 1 (Reference 3) (originally from Reference 4). Reference 3 indicates that the results due to IE for CASS materials (i.e., CF-8) are consistent with the overall fracture behavior of austenitic stainless steel weld material. Also, the data in MRP-276 for austenitic stainless steel

welds indicates consistency between the reduction in fracture toughness under both fast breeder and PWR conditions. Therefore, it is reasonable to assume that the fracture toughness behavior due to IE for CASS materials in PWR conditions (i.e., IMI guide tube spider castings) is consistent with the data for austenitic stainless steel welds in fast reactors. Per MRP-276, the results of the tests from the fast-breeder reactors indicate that 1) exposures up to 1 dpa have no significant effect on fracture resistance, 2) beyond 1 dpa, fracture resistance diminishes more rapidly than in the base metal because the embrittled delta-ferrite serves as an effective microvoid nucleation site, 3) neutron exposures of 2-4 dpa are seen to cause a 70-90% reduction in J_{IC}, and 4) at exposures above 10 dpa, saturation J_{IC} values range from 10-30 kJ/m^2.

Therefore, the following guidelines were used for the estimation methodology for the various locations of IMI guide tube spider castings, with interpolation between the various data points:

This same estimation methodology was used for all saturation percentages presented in Section 4.2.2 of ANP-3267, Revision 0.

References for RAI-4:

1. NUREG/CR-7027, "Degradation of LWR Core Internal Materials Due to Neutron Irradiation," December 2010, NRC ADAMS Accession Number ML102790482.

2. Kim, C., et. al., "Embrittlement of Cast Austenitic Stainless Steel Reactor Internals Components," Proc. 6th Intl. Symp. on Contribution of Materials Investigations to Improve the Safety and Performance of LWRs, Vol. 1, Fontevraud 6, French Nuclear Energy Society, SFEN, Fontevraud Royal Abbey, France, Sept. 18-22, 2006.

3. Materials Reliability Program: A Review of Radiation Embrittlement of Stainless Steels for PWRs (MRP-79) – Revision 1, EPRI, Palo Alto, CA: 2004. 1008204.

4. Mills, W.J., "Fracture Toughness of Irradiated Stainless Steel Alloys," Nuclear Technology, 82, 1988, pp. 290-303.

2.5 RAI-5

2.5.1 Statement of RAI-5

Section 4.2.3 describes the flow-induced vibration (FIV) analysis for the IMI guide tube spiders installed at the ONS units. Please address the following:

1. Provide the identifier and title of the report for the FIV analysis.

2. This section revealed that three configurations of the guide tube spider were adopted in the FIV analysis. [

   ] Please clarify the difference between these two configurations.

3. This section also states, "[

   ]" Please confirm that the reason for this is due to the fact that the high cycle excitation has a much higher frequency than the fundamental frequency of the guide tube with welded spiders such that the dynamic response of the spider casting will experience little change when the spider tube's natural frequencies decrease due to failure of one leg. Provide additional reasons to account for factors other than the above.

2.5.2 Response to RAI-5

2.5.2.1 *Response to Part 1:*

AREVA would prefer not to provide this information in this response because the document is AREVA Proprietary. However, the NRC is welcome to view the document through appropriate means.

2.5.2.2 *Response to Part 2:*

[ ]

2.5.2.3 *Response to Part 3:*

A response to this item will be provided by a later date.

2.6 RAI-6

2.6.1 Statement of RAI-6

Section 4.2.4 states that, "[

]" Provide the relevant stresses to support your statement. Also, provide the difference of the spider center displacement between the design configuration and the configuration with one failed leg, demonstrating that this displacement difference is not big enough to cause guide tube misalignment.

2.6.2 Response to RAI-6

The significant loading on the spider is the [

]

[

]

2.7 RAI-7

2.7.1 Statement of RAI-7

Section 5.2.2 mentioned that two vent valve retaining rings were replaced in 2012. Provide the root cause for these two failed retaining rings to demonstrate that your conclusion in Section 5.4, "[

]" is still valid.

2.7.2 Response to RAI-7

Replacement of the vent valve assembly was not due to aging degradation of the retaining rings. During a routine inspection of the internal vent valves in 2012 at the Oconee Nuclear Station (ONS) Unit 1, one of the eight vent valves was discovered to have a damaged right-hand jackscrew and areas of upset metal. The apparent cause of the damage was impact of the plenum with the top of the left jackscrew during plenum installation, which can occur if the plenum is not level during insertion. The vent valve retaining rings were removed along with the entire vent valve assembly and replaced with a spare vent valve. Due to the vent valve configuration, it is generally preferred to replace the entire assembly when damage is found rather than replace that part if an alternate assembly is readily available.

2.8 RAI-8

2.8.1 Statement of RAI-8

Section 5.2.2 states, "[ ]" Provide basis for this expectation.

2.8.2 Response to RAI-8

This value provided in the report is based on the value of [ ] presented in Table 4 of Reference 1 for a 17-4 PH material. This value is listed as "K_Q" or a conditional K value in the table; however the text describes this as being a valid K_{IC} value for the 17-4 PH material. The material is listed as having an austenitizing heat treatment of 1040°C (1904°F) and oil quenching, a tempering of 600°C (1112°F) for 4 hours followed by air cooling, and then aged for 5000 hours at 400°C (752°F).

[ ]

The material's susceptibility to TE is dependent on the material composition and tempering treatment.

[ ]

The 17-4 PH material in Reference 1 is shown to be fully saturated regarding TE after aging for 5000 hours at 400°C (752°F), based on a material with similar heat treatment parameters [5], [ ]

References for RAI-8:

1. Wagner, et. al., "Toughness and Fatigue Properties of Martensitic Stainless Steels for Nuclear Applications," 1984 ASM International Conference on New Developments in Stainless Steel Technology, Detroit, Michigan, American Society for Metals, Metal/Materials Technology Series.

2. E. Herny, "Mechanical Characterization and Investigation of Thermal and Thermomechanical Aging Mechanisms of Martensitic Stainless Steel 15-5 PH," International Heat Treatment and Surface Engineering, Volume 3, Number 1-2, Pages 65-69, 2009.

3. Tampigny, R., et. al., "In-Service Thermal Aging of Martensitic Stainless Steels," Paper Reference Number A128-T07, Fontevraud 7, September 2010.

4. Xu, H. and Fyfitch, S., "Aging Embrittlement Modeling of Type 17-4 PH at LWR Temperatures," Tenth International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, NACE, 2001.

5. Foct, F. et. al., "Thermal Aging and Stress Corrosion Cracking Resistance of Martensitic Stainless Steels used in High Temperature Water," Materiaux 2002.

6. Yrieix, B. and Guttmann, M., "Aging Between 300°C and 450°C of Wrought Martensitic 13-17 wt% Cr Stainless Steels," Materials Science and Technology, Volume 9, February 1993, Pages 125-134.

3.0 OVERALL REPORT REFERENCES

1 Duke Energy letter, ONS-2013-034 to the NRC dated November 29, 2013, "Duke Energy Carolinas, LLC, Oconee Nuclear Station Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287, Propose License Amendment Request for the Reactor Vessel Internals Inspection Plan; LAR Number 2010-06, Supplement 5, "ADAMS Accession Number ML13339A347.

2 Request for Additional Information (RAI) Related to the MRP-227-A Applicant/Licensee Action Item #7 Analysis Supporting the Reactor Vessel Internals Inspection Plan Amendment Request for Oconee Nuclear Station (ONS), Units 1, 2, and 3 (Received February 5th, 2014)

ATTACHMENT 2

AREVA AFFIDAVIT

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA Inc. (AREVA) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in licensing document ANP-3267Q1, Revision 0, entitled, "Response to NRC Request for Additional Information (RAI) Related to the MRP-227-A Applicant/Licensee Action Item #7 Analysis Supporting the Reactor Vessel Internals Inspection Plan Amendment Request for Oconee Nuclear Station (ONS), Units 1, 2, and 3," dated March 2014 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be

withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(c) and 6(d) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 5th

day of March, 2014.

Danita R. Kidd
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 12/31/16
Reg. # 205569

DANITA R. KIDD
COMMONWEALTH OF VIRGINIA
NOTARY PUBLIC