ATTACHMENT 1

AREVA Document No. ANP-3102Q1, "Response to NRC Request for Additional Information Regarding License Amendment Request to Update Pressure-Temperature Limit Curves for Three-Mile Island Unit 1," Revision 0, dated August 2013

AREVA

Response to NRC Request for Additional Information Regarding License Amendment Request to Update Pressure-Temperature Limit Curves for Three- Mile Island Unit 1

ANP-3102Q1
Revision 0

August, 2013

AREVA NP Inc.

Nature of Changes

Revision	Section(s) or Page(s)	Description and Justification
0	All	Initial Issue

Contents

Nomenclature

Acronym	**Definition**
ASTM	American Society of Testing and Materials
B&W	Babcock & Wilcox
CMTR	Certified Material Test Report
EFPY	Effective Full Power Years
EOL	End of License or End of Life
INF	Inlet Nozzle Forging
IS	Intermediate Shell
LAR	License Amendment Request
LLC	Limited Liability Company
LNBF	Lower Nozzle Belt Forging
LS	Lower Shell
MUR	Measurement Uncertainty Recapture uprate
NRC	United States Nuclear Regulatory Commission
OD	Outer Diameter
ONF	Outlet Nozzle Forging
P-T	Pressure-Temperature
RAI	Request for Additional Information
RCPB	Reactor Coolant Pressure Boundary
RT_{NDT}	Reference Temperature for Nil-Ductility Transition
RV	Reactor Vessel
TMI-1	Three Mile Island Unit 1
TS	Technical Specification
UNBF	Upper Nozzle Belt Forging
USE	Upper Shelf Energy

ABSTRACT

AREVA Document ANP-3102, Revisions 1 and 2, "Three-Mile Island Unit 1 Appendix G Pressure-Temperature Limits at 50.2 EFPY with MUR," were prepared by AREVA for Exelon Generation Company, LLC (Exelon). Subsequently AREVA Document ANP-3102, Revision 1 was inadvertently submitted to the NRC by Exelon with the associated license amendment request (LAR) to update the P-T limits.

The NRC has issued the first set of Requests for Additional Information (RAIs) on this submittal, and this report provides the responses for RAI 1 and RAI 2.

1.0 INTRODUCTION AND SUMMARY

AREVA Document ANP-3102, Revision 1[1] and 2 [2], “Three-Mile Island Unit 1 Appendix G Pressure-Temperature Limits at 50.2 EFPY with MUR,” were prepared by AREVA for Exelon Generation Company, LLC (Exelon). Subsequently AREVA Document ANP-3102, Revision 1 was inadvertently submitted to the NRC by Exelon[3]. The NRC has issued the first set of Requests for Additional Information (RAIs)[4] on this submittal, and this report provides the answers to those RAIs.

2.0 REQUESTS FOR ADDITIONAL INFORMATION (RAIs) AND RESPONSES

The NRC RAIs are reproduced from Reference 4 in Sections 2.1.1 through 2.2.1. The AREVA/Exelon responses are in Sections 2.1.2 through 2.2.2.

2.1 RAI 1

2.1.1 Statement of RAI 1

Issue: Section 4.6 of Attachment 4, AREVA Document No. ANP-3102, "Three-Mile Island, Unit 1 Appendix G Pressure-Temperature Limits at 50.2 EFPY [Effective Full-Power Years] with MUR [Measurement Uncertainty Recapture Uprate]," describes the reactor coolant temperature-time histories used in the calculations of the revised P-T limits, as specified below:

> The following input temperate-time histories are considered:
>
> > Normal Ramp Heatup, 50 °F/hr. [degrees Fahrenheit per hour]
> >
> > Normal Step Heatup, 50 °F/hr.
> >
> > Normal Ramp Cooldown, 100 °F/hr to 225 °F then 30 °F/hr to 70 °F.
> >
> > Normal Step Cooldown, 100 °F/hr to 225 °F then 30 °F/hr to 70 °F.

Note 2 to Figure 3.1-2 in the proposed TS markup, Attachment 2, shows the cooldown ramp history as follows:

> T > 255 °F 100 °F/hr or 15 °F / 9 min. Steps
>
> T > 255 °F 30 °F/hr or 15 °F / 30 min. Steps

Request: Explain this discrepancy, or revise the submittal to make the text consistent, either 255 °F or 225 °F, with the actual cooldown history that was used. Also, please clarify whether Attachment 4 is considered revision 1 or 2, because the attachment listing does not appear to match the attached document.

2.1.2 Response to RAI 1

The discrepancy of the time history descriptions between the contents of "*Section 4.6 of Attachment 4, AREVA Document No. ANP-3102*" and the "*Note 2 to Figure 3.1-2 in the proposed TS markup, Attachment 2*" is due to a typographical error in *AREVA Document No. ANP-3102*. The temperature at which the normal ramp and step cooldown rates change should have been noted in *AREVA Document No. ANP-3102* as 255 °F and not 225 °F. The cooldown ramp history in Note 2 of Figure 3.1-2 in the proposed TS markup, Attachment 2, is correct.

The text in *AREVA Document No. ANP-3102* was revised as follow:

> Normal Ramp Heatup, 50 °F/hr.
> Normal Step Heatup, 15 °F/ 18 min. steps
> Normal Ramp Cooldown, 100 °F/hr to 255 °F then 30 °F/hr to 70 °F.
> Normal Step Cooldown, 15 °F/ 9 min. steps to 255 °F then 15 °F/ 30 min. steps to 70 °F.

The correct revision of ANP-3102 is Revision 3, which is attached, and replaces Attachment 4 of the LAR. ANP-3102 is re-issued as ANP-3102, Rev. 3 to correct this typographical error. The results of the detailed P-T limits analyses summarized in *AREVA Document No. ANP-3102* used the correct temperature-time history inputs, and is consistent with the temperature-time history information provided in *Note 2 to Figure 3.1-2 in the proposed TS markup*. This change has no impact on the P-T limit curves provided in "*Attachment 4, AREVA Document No. ANP-3102*" and "*in the proposed TS markup, Attachment 2*".

Attachment 4 was intended to contain revision 2 but now contains revision 3 of ANP-3102. Revision 2 corrected the entry on the first column last row from "IS to LS Circ. Weld (63%)" to "LS Longit. Weld (OD 63%)" in both Tables 1 and 2. Revision 3 of ANP-3102 contains editorial changes on pages 2, 6, 13, and 53 as described on the record of revision page. These changes are editorial and do not have any impact on the results.

2.2 RAI 2

2.2.1 Statement of RAI 2

Background: P-T limit calculations for ferritic reactor coolant pressure boundary (RCPB) components that are not RV beltline shell materials, may define curves that are more limiting than those calculated for the RV beltline shell materials. This may be due to the following factors:

1. Some ferritic RCPB components that are not RV beltline shell materials, such as nozzles, penetrations, and other discontinuities, are complex geometry components that exhibit significantly higher stress intensities than those for the RV beltline region. These higher stresses can potentially result in more restrictive P-T limits, even if the RT_{NDT} for these components is not as high as that of RV beltline materials that have simpler geometries.

2. Ferritic RCPB components that are not RV beltline shell materials may have material properties, in particular initial RT_{NDT} values, which may define more restrictive P-T limits that those for the RV beltline shell materials.

Issue: In Attachment 4 to the submittal dated December 14, 2012, the licensee submitted information indicates that the beltline weld P-T curves are more conservative (limiting) than the P-T curves for the inlet and outlet nozzles. Attachment 4 also provided all of the inputs (per NRC Generic Letter 92-01) for evaluating the properties at the end of extended life for the beltline materials (chemistry and initial RT_{NDT} and upper shelf Charpy energy, as well as neutron fluence) used to generate the limiting beltline weld P-T curves. Attachment 4, section 4.7 indicates that P-T curves for the inlet and outlet nozzles were based on an assumed RT_{NDT} of 60 °F for the limiting nozzle, but does not provide any justification for this assumption. TMI-1 Updated Final Safety Analysis Report, Table 4.3-3, provides Charpy impact data for the nozzles, but no copper content. Neutron fluence values for the nozzle forgings are not included in any of the attachments to the submittal dated December 14, 2012.

Request: The NRC staff requests that the licensee provide the inputs for evaluating the properties at the end of extended life for the inlet and outlet nozzles - chemistry and initial unirradiated RT_{NDT}, unirradiated upper shelf Charpy energy, and fluence.

2.2.2 Response to RAI 2

2.2.2.1 *General Response*

For TMI-1, the inlet and outlet nozzle fluences are bounded by 3.01×10^{16} n/cm^2, (E > 1.0 MeV) (see Section 2.2.2.2). The attenuated fluences at the inlet and outlet nozzle corner flaw locations are even lower than this value. Per NUREG-1801, Revision 1 and Revision 2[5], a time-limited aging analysis for neutron irradiation embrittlement is required for materials with a neutron fluence greater than 1×10^{17} n/cm^2, (E > 1.0 MeV), therefore, embrittlement in this region does not need to be considered for the period of extended operation.

Certain properties (initial RT_{NDT}, copper content, upper shelf energy) were not measured for all materials during initial fabrication of the TMI-1 reactor pressure vessel. For beltline materials, archive materials were maintained to support the ASTM E185 reactor vessel surveillance program. The archive material permitted measurement of those properties for beltline materials when the importance of these properties became apparent. However, archive material was not maintained for the inlet and outlet nozzle forgings, since they were not considered limiting materials (in accordance with ASTM E185) in selection of the reactor vessel surveillance program materials. Thus, certain material properties are not available for the inlet and outlet nozzle forgings as described in the following section.

2.2.2.2 *Specific Response to Requested Inputs*

<u>RV Inlet and Outlet Nozzle Chemistries</u>

Per Regulatory Guide 1.99, Revision 2[6], the chemistry items required to evaluate the effect on properties due to irradiation are weight percent copper and nickel. Per the certified material test reports (CMTRs), the nickel contents are as follows: 0.74, 0.75, 0.68, and 0.80 wt% for the four inlet nozzle forgings and 0.70 and 0.70 wt% for the two outlet nozzle forgings. These nickel contents represent the higher of the ladle and check analyses. The copper content was not reported on the CMTRs.

RV Inlet and Outlet Nozzle Unirradiated Reference Temperature Nil-Ductility Temperature (RT_{NDT})

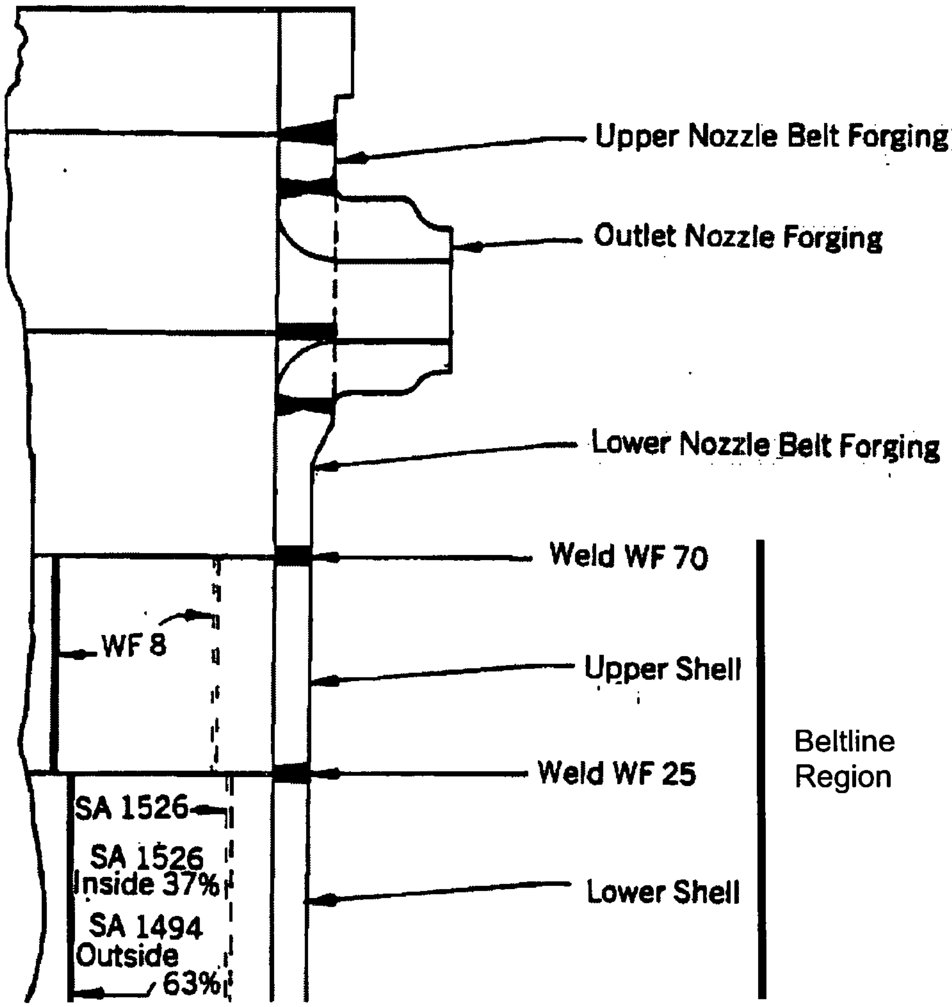
Per the NRC approved BAW-10046A, Rev. 2[7], the estimated unirradiated reference temperature nil-ductility temperature (RT_{NDT}) for the TMI-1 reactor vessel inlet and outlet nozzle forgings is 60 °F. Based on the fluence values discussed in Section 2.2.2.1, continued use of the unirradiated initial RT_{NDT} used in BAW-10046A, Rev. 2, is appropriate for deriving P-T limits for the Three Mile Island Unit 1 nozzle regions for 50.2 EFPY.

RV Inlet and Outlet Nozzle Unirradiated Upper Shelf Energy (USE)

The CMTRs for the TMI-1 inlet and outlet nozzle forgings report only 10 °F Charpy data, which is not sufficient to determine the USE per ASTM E185-82 (as directed by 10CFR50, Appendix G). The CMTRs show that the average Charpy impact energy at 10 °F is greater than 75 ft-lbs for each nozzle. Based on the fluence values discussed in Section 2.2.2.1, the Three Mile Island Unit 1 nozzle forgings will have adequate USE at 50.2 EFPY.

RV Inlet and Outlet Nozzle Fluences

Nozzle region fluence values for TMI-1 were provided as part of the TMI-1 (Docket No. 50-289) License Renewal response to RAI 4.2.0.0-01.[8] The specific TMI-1 nozzle region fluence values are listed in Table 2.4, page 11 of 35, of Enclosure A, Attachment 1, to AmerGen Letter #5928-08-20164.[8] The nozzle region locations (other than the Lower Nozzle Belt Forging (LNBF), which is connected to Weld WF-70) are not part of the beltline areas as defined by 10 CFR 50.61(a)(3) and the NUREG-1801 fluence value discussed in Section 2.2.2.1. The nozzle region locations relative to those beltline areas are shown in the following simplified sketch. Inlet and outlet nozzle centerlines are located on the same elevation. The diameter of the inlet nozzles are smaller than the diameter of the outlet nozzles.



The fluence values for the nozzle region are best estimate values generated based on the NRC-approved fluence analysis methodology in BAW-2241P-A[9], with no characterization of associated uncertainty.

Fluence values were provided in Attachment 1, Table 2.4 of 5928-08-20164[8] for 48 EFPY, 50 EFPY and 52 EFPY. The current estimated EOL based on license renewal is 50.2 EFPY, as described in the LAR. A simplified table of fluence values at 50 EFPY (from Table 2.4[8]), 50.2 EFPY (generated by linear interpolation), and 52 EFPY (from Table 2.4[8]) are shown below.

Reactor Vessel Location	**50 EFPY [n/cm^2]***	**50.2 EFPY [n/cm^2]***	**52 EFPY [n/cm^2]***
Flange to Upper Nozzle Belt Forging (UNBF)	7.36E+14	7.39E+14	7.66E+14
Lower Nozzle Belt Forging (LNBF) to Outlet Nozzle Forging (ONF) Weld	3.00E+16	3.01E+16	3.12E+16
LNBF **	1.65E+19	1.66E+19	1.71E+19

* Values are for internal, “wetted” surface of the reactor vessel. These fluence values are non-proprietary.

** Fluence is conservatively the same as the beltline circumferential weld, WF-70, that connects the LNBF to the upper shell.

Note the LNBF to ONF Weld fluence values conservatively bound the LNBF to Inlet Nozzle Forging (INF) Weld fluence values as the outlet diameter is larger and the LBNF to ONF weld is closer to the top of the core. There are more recent fluence calculations, but those values are bounded by those provided in the Reference 8 letter.

3.0 REFERENCES

1 ANP-3102, Revision 1, (AREVA Document ID 77-3102-001), "Three-Mile Island Unit 1 Appendix G Pressure-Temperature Limits at 50.2 EFPY with MUR," September, 2012.

2 ANP-3102, Revision 2, (AREVA Document ID 77-3102-002), "Three-Mile Island Unit 1 Appendix G Pressure-Temperature Limits at 50.2 EFPY with MUR," September, 2012.

3 Exelon Generation Company, LLC (Exelon), Three Mile Island Nuclear Station, Unit 1, Docket Number 50-289, "Revision to the Pressure and Temperature Limit Curves and the Low Temperature Overpressure Protection Limits, and 10 CFR 50.12 Exemption Request - Initial RT_{NDT} Values for Linde 80 Welds," Letter Number TMI-12-183, NRC ADAMS Accession Number ML12353A319, December 14, 2012.

4 USNRC Letter, (Exelon), "Three Mile Island Nuclear Station, Unit 1 – Request for Additional Information Regarding Proposed Revision to Pressure and Temperature Limit Curves and Exemption Request For Initial Reference Temperature Values," Docket No. 50-289, NRC ADAMS Accession Number ML13193A175, July 22, 2013.

5 NUREG-1801, Rev. 2, "Generic Aging Lessons Learned (GALL) Report," December, 2010.

6 U. S. Nuclear Regulatory Commission, "Radiation Embrittlement of Reactor Vessel Materials," Regulatory Guide 1.99, Revision 2, May, 1988.

7 BAW-10046A, Rev. 2, Topical Report, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," June, 1986.

8 Exelon Correspondence 5928-08-20164, September 10, 2008, Enclosure A, Attachment 1: AREVA Report 86-9038511-000, "TMI License Renewal RPV Final Fluence Report," dated March 15, 2007, NRC ADAMS Accession Number ML082560178.

9 BAW-2241P-A, Rev. 2, Topical Report, "Fluence and Uncertainty Methodologies".