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June 18, 1997

2CAN069709

U. S. Nuclear Regulatory Commission
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Washington, DC 20555

Subject: Response to Generic Letter 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity," for ANO-2
TAC Nos. M92642 and M77399

Gentlemen:

By letters dated August 11, 1995 (0CAN089505) and November 13, 1995 (0CAN119502), Entergy Operations provided the information for Arkansas Nuclear One (ANO) requested by Generic Letter 92-01, Revision 1, Supplement 1. The required six month response to parts (2), (3), and (4) for ANO-2 were to be provided subsequent to completion of further data review. The response to parts (2), (3), and (4) of the generic letter for ANO-2 is attached.

By letter dated March 7, 1997 (2CNA039701), the Staff transmitted Amendment Number 180 to the ANO-2 Technical Specifications regarding low temperature overpressure protection (LTOP). The Staff stated that the assumption for the 1/4T fluence at 21 effective full power years (EFPYs) is conservative for several reasons. One of which was that "the value for 21 EFPYs will be used only to the next refueling when the plant will have experienced less than 13 EFPYs." This is not consistent with the information that was submitted by Entergy Operations in correspondence dated September 5, 1996 (2CAN099603) in which Entergy Operations committed to provide a best estimate of remaining EFPY margin, the next vessel specimen withdrawal schedule, and when a revised pressure-temperature (P-T) analysis would be performed. The consideration of the adequacy of the ANO-2 P-T curves "only to the next refueling" was not discussed in the September 5, 1996, submittal. The committed information is provided in the attachment to this submittal. Based on the information provided in the attachment, Entergy Operations concludes that the ANO-2 Technical Specification reactor vessel specimen withdrawal schedule, P-T limits, and LTOP limits do not need to be revised prior to reaching 21 EFPY.

Should you have any questions, please contact me.

A028

Very truly yours,

Dwight C. Mims
Director, Licensing

DCM/nbm
Attachment

To the best of my knowledge and belief, the statements contained in this submittal are true.

SUBSCRIBED AND SWORN TO before me, a Notary Public in and for Johnson **County and the State of Arkansas, this** 18 **day of** June**, 1997.**

Juana M. Tapp
Notary Public
My Commission Expires 11-8-2000

OFFICIAL SEAL
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ANO-2 GL 92-01 Revision 1, Supplement 1 Report

1.0 Background/Purpose

The purpose of Reference 1 was to require licensees to identify, collect and report any new data pertinent to the analysis of the structural integrity of their reactor pressure vessels (RPVs) and to assess the impact of that data on their RPV integrity analyses relative to the requirements of 10CFR50.60, 10CFR50.61, Appendices G and H of 10CFR50, and any potential impact on low temperature overpressure (LTOP) limits or pressure/temperature (P/T) limits.

To fulfill the purpose of the supplement, the NRC has required licensees to provide the following information:

1. A description of those actions taken or planned to locate all data relevant to the determination of RPV integrity, or an explanation of why the existing data base is considered complete as previously submitted;

2. An assessment of any change in best-estimate chemistry based on consideration of all relevant data;

3. A determination of the need for use of the ratio procedure in accordance with the established Position 2.1 of Regulatory Guide 1.99, Revision 2, for those licensees that use surveillance data to provide a basis for the RPV integrity evaluation; and

4. A written report providing any newly acquired data as specified above and (1) the results of any necessary revisions to the evaluation of RPV integrity in accordance with the requirements of 10CFR50.60, 10CFR50.61, Appendices G and H to 10CFR50, and any potential impact on the LTOP or P/T limits in the technical specification or (2) a certification that previously submitted evaluations remain valid. Revised evaluations and certifications should include consideration of Position 2.1 of Regulatory Guide 1.99, Revision 2, as applicable, and any new data.

The information for Item 1 above was required to be submitted to the Staff within 90 days from the date of Reference 1. Within six months from the date of issuance of Reference 1, a written response to Items 2, 3, and 4 was required. The information requested in Item 1 was submitted to the NRC via Reference 2.

The purpose of this report is to compile all the required information to address items 2, 3, and 4 of Reference 1 for Arkansas Nuclear One, Unit 2 (ANO-2) into one document. For completeness, the information that was requested in Item 1 is repeated in this report. Each of the items above will be addressed individually.

2.0 References

1. Generic Letter (GL) 92-01, Revision 1, Supplement 1: "Reactor Vessel Structural Integrity," issued on May 19, 1995

2. ANO Letter 0CAN089505, "90 Day Response to Generic Letter 92-01, Revision 1, Supplement 1, 'Reactor Vessel Structural Integrity'," dated August 11, 1995, from D. C. Mims (ANO) to the U. S. NRC Document Control Desk

3. ANO Letter 0CAN119502, "6 Month Response to Generic Letter 92-01, Revision 1, Supplement 1, 'Reactor Vessel Structural Integrity'," dated November 13, 1995, from D. C. Mims (ANO) to the U. S. NRC Document Control Desk

4. ABB-CE Report A-PENG-ER-002, Revision 0, "The Reactor Vessel Group Records Evaluation Program Phase II Final Report for the ANO-2 Reactor Pressure Vessel Plates, Forgings, Welds, and Cladding," October 1995

5. ABB-CE Report CE NPSD-1039, "Best Estimate Copper and Nickel Values in CE Fabricated Reactor Vessel Welds," December 1996

6. Battelle Report, "Final Report on Examination, Testing, and Evaluation of Irradiated Pressure Vessel Surveillance Specimens from the Arkansas Nuclear One Unit 2 Generating Plant to Arkansas Power and Light Company, " by Lowry, Landow, Failey, Jung, Manahan, and Denning, May 1, 1984

7. ABB-CE Report CEN-15(A), "Summary Report on Manufacture of Test Specimens and Assembly of Capsules for Irradiation Surveillance of Arkansas Nuclear One - Unit 2 Reactor Vessel Materials," May 30, 1975

8. ANO Letter 2CAN069109, "Proposed Change to the Technical Specification Pressure/Temperature Limits," dated June 18, 1991, from N. S. Carns (ANO) to the U. S. NRC Document Control Desk

9. ABB-CE Inter-Office Correspondence to Mr. B. R. Moss from J. A. Kosik, Metallurgical R&D Department Chattanooga, Arkansas Power & Light Surveillance Test Contract 73170 Job No. B-38071 960001, dated January 30, 1973

10. ABB-CE Inter-Office Correspondence to Mr. B. R. Moss from J. A. Kosik, Metallurgical and Materials Department Chattanooga, Surveillance Program Contract 73170 Job No. B-38071, Project No. 960001, dated September 3, 1974

11. U. S. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"

12. ABB-CE Report MCC-91-097, "RT_{NDT} of Arkansas Nuclear One - Unit 2 Reactor Vessel Materials (611904), from S. T. Bryne to C. D. Stewart, dated February 21, 1991

13. Calculation 90-E [illegible]97-01, Revision 1, "Reactor Vessel Fluence Determination"

14. ABB-CE Report A-MPS-ER-002, "Final Report on Reactor Vessel Appendix G Pressure - Temperature Limits for Arkansas Nuclear One Unit 2 for 21 Effective Full Power Years," May 1991

15. SSI Calculation #278-02.6, "Generic Letter 88-11 Analysis Results for ANO-1 and ANO-2," January 1989

16. ANO-2 Technical Specification 3/4.4.9, "Pressure / Temperature Limits," Amendment 124

17. 10CFR50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

18. Report 96-R-2030-02, Revision 0, "Revised Reactor Vessel Fluence Determination"

19. Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence"

20. NRC letter dated January 2, 1996, "Updated Values for Pressured Thermal Shock Reference Temperatures - Calvert Cliffs Nuclear Power Plant, Units Nos. 1 and 2," D. G. McDonald, Jr. (NRC) to R. E. Denton (Baltimore Gas and Electric)

3.0 Evaluation

Each of the NRC's requests for information is addressed below.

3.1 <u>Description of those actions taken to locate all data relevant to the determination of RPV integrity</u>

In Reference 2, the NRC was informed that in December 1992, Entergy Operations, representing ANO-2, joined the Combustion Engineering Reactor Vessel Group, now the Combustion Engineering Owners Group (CEOG) Reactor Vessel Working Group (RVWG).

This group initiated a project, "Design and Fabrication Records Evaluation Program," to evaluate the design and fabrication records for Combustion Engineering reactor vessels. One primary task involved with this project was to assemble and evaluate the original equipment manufacturer (OEM) design and fabrication records. The purpose was to identify and ensure the retention of valuable information in an organized and easily retrievable form.

Reference 4 provides a vessel specific summary of the evaluated fabrication records for the ANO-2 reactor vessel. This report applies to the ANO-2 reactor vessel portion of the global data matrix that was developed as part of the above mentioned program. Included in the records evaluated for Reference 4 were the original equipment manufacturing records in possession of ABB/Combustion Engineering. Reference 5 addresses the best estimate copper and nickel content in reactor pressure vessel welds fabricated by Combustion Engineering. It describes the processes used to establish data pedigree and to analyze the copper and nickel content data. It presents the evaluated database and the best estimate Cu and Ni contents for weld electrode heats used in fabricating reactor vessel beltline welds.

The database used for the determination of the best-estimate chemistry for the beltline welds consists of 1,881 entries obtained from the Combustion Engineering Chattanooga weld deposit log book, the Oak Ridge National Labs PR-EDB (Version 2), a CEOG evaluation, a Baltimore Gas and Electric evaluation, a Consumers Power Company evaluation, Electric Power Research Institute RPVDATA (Version 1.2), and from miscellaneous other sources.

3.2 An assessment of any change in best-estimate chemistry based on consideration of all relevant data

3.2.1 Beltline Plates

The copper and nickel data for the six ANO-2 reactor vessel beltline plates is obtained from the RVG report, Reference 4, and is listed below in Table 3.2.1-1. There are four different source categories in Table 3.2.1-1. The "Surv" values are results of analyses done at the Combustion Engineering Chattanooga facilities directly in support of the ANO-2 reactor vessel surveillance program. The "Code" values are results of analyses done at the Chattanooga facilities for the vessel beltline materials to meet ASME Code requirements. The two "Lukens" values are results of analyses done at Lukens Steel Company facilities as a ladle and check analysis to meet Combustion Engineering purchase specifications.

TABLE 3.2.1-1
Reactor Vessel Beltline Plate Cu and Ni Content

Plate Code:	C8009-1	C8009-2	C8009-3	C8010-1	C8010-2	C8010-3
Heat #:	C8161-3	C8161-1	C8182-2	C8161-2	B2545-1	B2545-2

Copper (Cu) Content

Surv. %	0.12	0.08	0.08	0.08	0.07	0.07
Code %	0.08	0.08	0.13	0.08	0.08	0.07
Lukens %	0.09	0.09	0.14	0.09	0.09	0.09
Lukens %	0.10	0.09	0.13	0.09	0.09	0.09

Nickel (Ni) Content

Surv. %	0.63	0.59	0.60	0.59	0.66	0.65
Code %	0.65	0.65	0.67	0.61	0.68	0.63
Lukens %	0.56	0.56	0.60	0.58	0.66	0.66
Lukens %	0.58	0.60	0.58	0.56	0.67	0.67

In Table 3.2.1-1, Plate C8009-3 shows large differences in measured copper content relative to the other plates. Plate C8009-3 was from a unique heat such that no direct comparisons can be made within the heat; however, the four copper measurements are very inconsistent. Plate C8009-3 is also the surveillance program plate. There are five specimens from the surveillance base metal, which was analyzed by Battelle Memorial Institute during testing of the first surveillance capsule, Reference 6. A summary of the copper and nickel content for these specimens is provided in Reference 6 and reiterated in Table 3.2.1-2. The fabrication report, Reference 7, and drawing E-6370-165-110 Rev. 02, provides justification that the five specimens are from the C-8009-3 plate.

TABLE 3.2.1-2
Cu/Ni Content of Surveillance Specimen

C8009-3 Specimen	% Cu	% Ni
13T Base (L)	0.078	0.559
14M Base (L)	0.077	0.548
21U Base (L)	0.075	0.545
264 Base (T)	0.078	0.555
257 Base (T)	0.078	0.559

The data in Tables 3.2.1-1 and 3.2.1-2 are used to establish the best-estimate values of copper and nickel for the beltline plates. Determining the "best-estimate" values is defined in 10CFR50.61, paragraph (b)2.iv, as the mean of measured values for a plate. The revised best estimate values and the corresponding chemistry factors for each plate are listed in Table 3.2.1-3. The best estimates are the arithmetic averages of the four measurements for each plate given in Table 3.2.1-1, except for C8009-3. The best estimate copper and nickel content for C8009-3 is the arithmetic average of the nine sets of measurements listed in Tables 3.2.1-1 and 3.2.1-2. Table 3.2.1-3 also has the "original" best-estimate chemistry values that were previously reported to the NRC (Reference 8).

TABLE 3.2.1-3
Comparison of Original to Revised Best-Estimate Cu/Ni Content of Reactor Vessel Beltline Plates

Plate Code	Original Best-Estimate Chemistry Cu %	Original Best-Estimate Chemistry Ni %	Revised Best-Estimate Chemistry Cu %	Revised Best-Estimate Chemistry Ni %
C8009-1	0.12	0.63	0.098	0.605
C8009-2	0.08	0.59	0.085	0.600
C8009-3	0.08	0.60	0.096	0.580
C8010-1	0.08	0.59	0.085	0.585
C8010-2	0.07	0.66	0.083	0.668
C8010-3	0.07	0.65	0.080	0.653

A chemistry assessment was first conducted for all six of the beltline plates. Reference 4 was used to obtain the various chemistry data for the six beltline plates. Reference 9 and 10 verified the chemistry for the unirradiated surveillance specimens.

Reference 4 also indicates that there are no "sister vessels" for the six reactor vessel beltline plates in ANO-2. Thus there will not be any additional sources of chemistry data for these plates except for the ANO-2 surveillance program.

3.2.2 Beltline Welds

There are three unique heats of welds present in the beltline region of the ANO-2 vessel. Table 3.2.2-1 presents the revised chemistry data for these welds, based on Reference 5. Table 3.2.2-1 also provides the chemistry data for these welds that have previously been submitted to the NRC (Reference 8).

TABLE 3.2.2-1
Comparison of Original to Revised Best-Estimate Cu/Ni Content of Reactor Vessel Beltline Welds

Component	Identification Number	Heat Number Identification	Original Best-Estimate Cu (%)	Original Best-Estimate Ni (%)	Revised Best-Estimate Cu (%)	Revised Best-Estimate Ni (%)
Intermediate Shell Longitudinal Welds	2-203-A, B, C	10120	0.05	0.18	0.046	0.082
Lower Shell Longitudinal Welds	3-203-A, B, C	10120	0.05	0.18	0.046	0.082
Upper / Intermediate Shell Girth Weld	8-203	10137	0.23	0.18	0.216	0.043
Lower / Intermediate Shell Girth Weld	9-203	83650	0.05	0.08	0.045	0.087

3.3 Determination of the need for the use of the ratio procedure

The ratio procedure is defined in Position 2.1, "Adjusted Reference Temperature," of Reference 11. This position states,

> "The adjusted reference temperature should be obtained as follows. First, if there is clear evidence that the copper or nickel content of the surveillance weld differs from that of the vessel weld, i.e., differs from the average for the weld wire heat number associated with the vessel weld and the surveillance weld, the measured values of ΔRT_{NDT} should be adjusted by multiplying them by the ratio of the chemistry factor for the vessel weld to that for the surveillance weld. Second, the surveillance data should be fitted using Equation 2 ($\Delta RT_{NDT} = (CF)f^{(0.28 - 0.10\log f)}$) to obtain the relationship of ΔRT_{NDT} to fluence. To do so, calculate the chemistry factor, CF, for the best fit by multiplying each adjusted ΔRT_{NDT} by its corresponding fluence factor, summing the products, and dividing by the sum of the squares of the fluence factors."

It should be noted that Position 2 and its subsections is applicable "[w]hen two or more credible surveillance data sets become available from the reactor in question." (Reference 11)

To date, only one surveillance data set has been examined. This data set was the 97° surveillance capsule. Battelle Columbus Laboratories issued a report on their examination, testing and evaluation of the specimen on May 1, 1984 (Reference 6). In addition there are no "sister vessels" for the six beltline plates in ANO-2 (Reference 4).

Based on the above, this regulatory position is not applicable to ANO-2. Therefore, the adjusted reference temperatures will be calculated using Position 1.1 of Reference 11.

3.4 Newly acquired data and results of RPV integrity evaluations

Sections 3.2.1 and 3.2.2 provide the data that was utilized in evaluating the best-estimate chemistry properties of the vessel beltline components. As can be seen the chemistry properties have been revised based upon the new databases. Therefore, several vessel evaluations were performed to identify the impact on current limits. These evaluations include the determination of the limiting component, past operability with the original P/T curves using the revised data, validating the current P/T and LTOP limits, determining the revised RT_{PTS}, and the decrease in the upper shelf energy.

3.4.1 Limiting Component Determination

The current P/T and LTOP limits are based on plate C8009-1 being the limiting component. Two classes of components, plates and welds, were reviewed in this evaluation. The limiting component is the one with the highest adjusted reference temperature (ART) value at the end of the period under consideration at the quarter and three-quarter vessel thickness (1/4t and 3/4t) locations.

The ART values were calculated in accordance with Reference 11, part C, section 1.1. The initial RT_{NDT} and the margin terms used in this methodology have not been revised due to the revision of the chemistry data. These terms are not impacted by this work.

3.4.1.1 Beltline Plates

Using the best-estimate chemistry values provided in Section 3.2 and Table 2 of Reference 11, a chemistry factor (CF) was obtained for each plate.

The vessel wall thickness (plate thickness), t, is 7.875 inches (cladding ignored), Reference 12. Adjusted reference temperatures are calculated at 1/4t and 3/4t. The neutron fluence factor was calculated according to Reference 11. The reactor vessel inner surface fluence per EFPY is 1.78 E + 18 n/cm^2 for all the beltline components (Reference 13) except for the upper to intermediate shell girth weld. The fluence rate for this weld is 0.513 E + 18 n/cm^2 (Reference 13). These values are consistent with the current 21 EFPY technical specification pressure/temperature limits.

For vessel heat-up and cool-down curves, the limiting beltline plate has the highest ART value at the end of the period under consideration at the one-quarter thickness (1/4t) or the three-quarter thickness (3/4t) location in the vessel. For the time period, 21 EFPY, the predicted limiting beltline plate is C-8010-1. Refer to Table 3.4.1-1.

TABLE 3.4.1-1
1/4t and 3/4t ART Values Used for 21 EFPY Limits

PLATE NUMBER	1/4 t ART Value	3/4 t ART Value
C8009-1	86.2	69.9
C8009-2	101.0	87.0
C8009-3	110.4	94.5
C8010-1	113.0	99.0
C8010-2	71.2	57.6
C8010-3	66.7	53.6

3.4.1.2 Beltline Welds

The ART values for the welds were not explicitly determined in this evaluation. Since the only term in the calculation of the ART that has been revised is the chemistry factor, if it can be shown that the revised chemistry factor for each of the welds is equal to or less than the chemistry factor used in Reference 12, the vessel plates remain the limiting component.

The value for the chemistry factor for each beltline weld from Reference 12 is provided in Table 3.4.1-2. Based upon the revised chemistry data for the welds (Table 3.2 2-1), revised chemistry factors were determined for each weld. This revised chemistry factor is provided in Table 3.4.1-2 as well.

There is one other weld in this region. This is the upper to intermediate shell, 8-203 (Heat 10137). This weld is not included in this evaluation since the fluence to this weld is lower than any of the other components (Reference 13).

Table 3.4.1-2
Comparison of Chemistry Factors for the Reactor Vessel Beltline Welds

Component	Identification Number	Heat Number Identification	Original Chemistry Factor	Revised Chemistry Factor
Intermediate Shell Long. Welds	2-203-A, B, C	10120	47	34
Lower Shell Long. Welds	3-203-A, B, C	10120	47	34
Inter. / Lower Shell Girth Weld	9-203	83650	35	34.1

It should be noted the chemistry factors in Table 3.4.1-2 were calculated using Table 1 of Reference 11.

It can be seen from Table 3.4.1-2 that the revised chemistry factors for the welds are lower than the original chemistry factors. As such, the welds are not the limiting component of the reactor vessel.

Based on the above discussions, the limiting component for the beltline components has been revised. It was plate C8009-1. Due to the changes in the chemistry data, the limiting component is now plate C8010-1. This change in limiting components requires additional evaluations which are presented in subsequent sections.

3.4.2 Past Operability

This evaluation provides a qualitative assessment of past operability with the original P/T curves using the "new" chemistry data and the best estimate fluence. The original Figure 3.4-2 of the ANO-2 Technical Specifications, was entitled "Reactor Coolant System Temperature Limitations for 0 to 10 Years of Full Power Operation." This is equivalent to 8.24 effective full power years (EFPY) of operation (10 years of 2900 MW_t operation at an 80% capacity factor; actual thermal rating of only 2815 MW_t). The current technical specification limits were approved by the NRC at approximately the same time the original limits expired.

On July 12, 1988, the NRC issued Generic Letter 88-11 which apprised licensees of the NRC's intent to use Reference 11, in reviewing submittals regarding pressure / temperature (P/T) limits and for analysis of the embrittlement of reactor vessel belt line materials. The methodology of Reference 11 was applied to the ANO-2 Technical Specification P/T limits in order to assess the impact of the revision of the regulatory guide.

The use of the Reference 11 methodology did not appear to require a modification of the technical specification limits. The shift in RT_{NDT} for 8 EFPY, was determined to be 4°F at the 1/4 vessel thickness location. This value and others were used to estimate that the resultant P/T limit curve would change by approximately 4°F from their then current values, indicating that changes were not required to be made in the 8.24 EFPY limits (Reference 15).

Based on the above, the Reference 11 methodology was used to calculate the adjusted reference temperature for each of the six plates for 8.24 EFPY. Two cases were performed. The first case will be with the appropriate original chemistry values for the plates. The second case will be with the revised chemistry values. The limiting values for both cases will then be compared to determine "past operability."

The original copper and nickel content of each plate was taken from Amendment 36 (dated March 31, 1976) of the final safety analysis report. Table 3.4.2-1 presents this data.

TABLE 3.4.2-1
Original Cu/Ni Content of Reactor Vessel Beltline Plates

Plate ID	Cu Content (%)	Ni Content (%)
C8009-1	0.12	0.63
C8009-2	0.08	0.59
C8009-3	0.08	0.60
C8010-1	0.08	0.59
C8010-2	0.07	0.66
C8010-3	0.07	0.65

Table 3.4.2-2 provides the revised copper and nickel content for the plates from Table 3.2.1-3 for ease of review.

TABLE 3.4.2-2
Revised Cu/Ni Content of Reactor Vessel Beltline Plates

Plate ID	Cu Content (%)	Ni Content (%)
C8009-1	0.098	0.605
C8009-2	0.085	0.600
C8009-3	0.096	0.580
C8010-1	0.085	0.585
C8010-2	0.083	0.668
C8010-3	0.080	0.653

Based on the information provided in Tables 3.4.2-1 and 3.4.2-2, the chemistry factor can be determined using Table 2 of Reference 11. Table 3.4.2-3 below provides the chemistry factor for each plate.

TABLE 3.4.2-3
Comparison of Chemistry Factors for the
Reactor Vessel Beltline Plates

Plate ID	Original CF	Revised CF
C8009-1	83.5	63.6
C8009-2	51	54.5
C8009-3	51	62.2
C8010-1	51	54.5
C8010-2	44	53.1
C8010-3	44	51.0

These values are input into the Reference 11 methodology to determine the ARTs.

It is assumed the initial RT_{NDT} and margin terms for each plate are the same for both cases.

The peak fluence at the end of 32 EFPY, cited in the final safety analysis report, is $3.47 * 10^{19}$ n/cm^2. Ratioing this to 8.24 EFPY, the peak inside surface fluence is then assumed to be $0.89 * 10^{19}$ n/cm^2. The 1/4t and 3/4t location fluences can then be determined by using equation 3 of Reference 11.

The 1/4t fluence value is $0.55 * 10^{19}$ n/cm^2. The 3/4t fluence value is $0.22 * 10^{19}$ n/cm^2. These values are used to determine both the original and revised ART for each plate.

Using the information provided above, the methodology described in Reference 11 was used to determine the ART value for each plate at 8.24 EFPY. Table 3.4.2-4 provides the ART values at the 1/4t and 3/4t locations using the original chemistry data. Table 3.4.2-5 provides the same information using the revised chemistry data.

TABLE 3.4.2-4
8.24 EFPY ART Values Using Original Chemistry Values

Plate ID	1/4t	3/4t
C8009-1	77.7	57.1
C8009-2	76.6	64.0
C8009-3	76.6	64.0
C8010-1	88.6	76.0
C8010-2	42.8	31.9
C8010-3	40.8	29.9

Plate C8010-1 is the limiting plate for both the 1/4t and 3/4t location.

TABLE 3.4.2-5
8.24 EFPY ART Values Using Revised Chemistry Values

Plate ID	1/4t	3/4t
C8009-1	60.8	44.9
C8009-2	79.5	66.0
C8009-3	85.6	70.1
C8010-1	91.5	78.0
C8010-2	50.3	37.1
C8010-3	46.3	33.6

Using the revised chemistry data, Plate C8010-1 remains the limiting plate for both locations. Note the ART for both locations for this plate has increased by 2 or 3°F using the revised chemistry data. This shift in the ART values has a very small impact on the pressure/temperature limits that were in place at the time. This minimal impact could have been handled within the uncertainties used in the analysis at the time.

It appears that the original technical specification limits were developed prior to the issuance of Regulatory Guide 1.99. Therefore the approach taken above is conservative. The original bases of the technical specification states

> "The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E > 1 MeV) irradiation will cause an

increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence can be predicted using Figure B 3/4.4-1. The heatup and cooldown limit curves shown on Figure 3.4-2 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments."

Using Figure B 3/4.4-1 and the fluence value of $0.89 * 10^{19}$ n/cm^2, a shift in RT_{NDT} of ~110°F can be determined. The values reported in Table 3.4.2-5 includes the initial RT_{NDT}, a margin term, and the shift in RT_{NDT}. It can be seen the shift calculated in accordance with the bases of the original technical specification is larger than the ART calculated using the Regulatory Guide 1.99, Revision 2 methodology.

Based on the above arguments, the ANO-2 reactor vessel remained operable in the past when the new chemistry data is accounted for.

3.4.3 Current Limits

According to the bases for the current P/T limits (Reference 16), the ART for 21 EFPY at the 1/4t position is 111°F and 96°F at the 3/4t location. These values are based on a vessel inner surface fluence of 3.74 E + 19 n/cm^2; the 1/4t fluence is 2.33 E + 19 n/cm^2; and the fluence at the 3/4t location is 9.06 E + 18 n/cm^2 (E > 1 MeV).

In the evaluation described in section 3.4.1, the new chemistry data and the fluence assumptions used in the current P/T limits were used. The ARTs that are calculated exceed the ART values provided in the basis of Reference 16. (See Table 3.4.1-1)

Based on this information, the period of applicability would be required to be revised from 21 EFPY to ~17 EFPY. This poses a concern with the next scheduled surveillance capsule withdrawal. The next capsule is scheduled to be withdrawn at 19 EFPY (Reference 16). With the required one year time frame to analyze and report the results of the capsule (Reference 17), there would be approximately one year to revise the P/T curves and gain NRC approval for the revised limits prior to the expiration of the current limits.

There are several options for the resolution of this issue. These options include revising the period of applicability of the current limits and the surveillance schedule now, revising the period of applicability of the current limits and leaving the surveillance schedule as it is; or evaluate the underlying bases for the current limits and remove unnecessary conservatisms from them.

The fluence estimates used to date are very conservative in nature. They are based on the one specimen capsule pulled at 1.69 EFPY and linearly extrapolated to 21 EFPY (Reference 13). The fuel management at the time the capsule was withdrawn and analyzed was a high leakage core design (Cycle 2). The ANO-2 fuel management went to a low leakage design in Cycle 6. The fuel management for the unit has remained with the low leakage design since that time. In addition, since the time the capsule was evaluated, ANO-2 has lowered the RCS inlet temperature due to other concerns. These two issues had a beneficial impact by reducing the flux of high energy neutrons to the vessel wall. However, the fluence estimates were not revised.

The fluence estimates were revisited to take advantage of some of the conservatisms in the simple linear extrapolation. The method used in the evaluation of the fluence included the use of the "excore ratios" that are determined as part of each cycle's startup predictions. The "excore ratio" is the ratio of a cycle's beginning of cycle (BOC) 100% power flux at the excore detector locations to the previous cycle's BOC 100% power flux at the same location. These ratios provide an indication of the amount of leakage from the core from cycle to cycle. It is assumed that the ratio of the fast flux from one cycle to the next is equal to this ratio.

There are 11 assemblies used in the calculation of the ratio. Assembly weighting factors are applied to the radial relative power density (RPDs) to determine a particular assembly's contribution to the response of the excore detector.

To determine a cycle's flux, the previous cycle's flux estimate is multiplied by the excore ratio for the cycle in question. For this methodology to work, the flux for Cycle 1 was determined. This was done by using the information from the capsule that was removed at the end of Cycle 2. The maximum surface fluence for the capsule was $3.01 \text{ E} + 18 \text{ n/cm}^2$.

This determination utilized the ENDF/B-IV cross-sectional library. In Reference 19, the NRC has noted that ENDF/B-IV libraries may underpredict the fluence the vessel wall is seeing due to an error in the Iron Inelastic Scattering cross-section. The ENDF/B-VI libraries corrected this deficiency. This underprediction could be significant (20 to 30%) according to the NRC.

In a letter to Baltimore Gas and Electric, dated January 2, 1996 (Reference 20), the NRC stated the following concerning projected neutron fluence,

> "The methodology employed the CASK cross section set. CASK is based in an early ENDF/B version which is known to have an iron scattering cross section error, which is corrected in ENDF/B-VI. However, we know from experience that this error appears only during neutron transmission through significant amounts of iron, as for example the thermal shield or

> the vessel. Neither of the Calvert Cliffs units is equipped with a thermal shield; thus, the staff does not expect the results to have been affected by the use of the CASK cross sections."

ANO-2 does not have a thermal shield; therefore based on the above, it is not expected that the maximum surface fluence would change if the analysis was reperformed using the ENDF/B-VI libraries.

There are two additional issues associated with the approach used that may impact the determination of the initial flux. These issues are the use of BOC RPDs versus end of cycle RPDs and the use of the RPDs from the reload reports versus as-built RPD data. Each of these issues are discussed below.

As a cycle progresses, each assembly's RPD changes as the power shifts from the center of the core to the periphery of the core. The excore ratio methodology was developed for the beginning of a cycle so the excores could be calibrated. The excores are periodically calibrated throughout the cycle. The shift in the RPDs from the beginning of a cycle to the end of a cycle is relatively small in magnitude.

The RPD information provided in each cycle's reload report is based on predictions for that cycle. The ratio that is provided is based on as-built data. It has been demonstrated that the ratio calculated using the reload report predicted values versus the ratio using the as-built data are very close.

To address all three issues listed above (ENDF/B-IV versus B-VI, power shift, and the use of reload report predicted RPDs), the 1.69 EFPY calculated fluence was increased by 10%.

Based on this methodology, the 21 EFPY inner surface fluence estimate is 2.94 E + 19 n/cm^2 which is lower than the value used in the bases of the current limits. Based on this new fluence estimate the 1/4t and 3/4t fluence was determined. Again these values were less than the ones used in the bases of the current limits.

Based on these new fluence values and the new chemistry values the ART for the 1/4t and 3/4t locations at 21 EFPY were determined. The 1/4t location ART was calculated to be 109.5°F which is less than the 111°F listed in the bases of the current limits. The revised 3/4t ART is 95.3°F as compared to the 96°F listed in the bases. Reference 18 provides the details of the evaluation described above.

Based on the above information, the limits currently listed in the ANO-2 Technical Specification are still applicable and the period of applicability for the limits can remain at 21 EFPY and the surveillance schedule does not need to be revised.

3.4.4 RT_{PTS}

10CFR50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," provides the methodology to calculate the reference temperature for the pressurized thermal shock (RT_{PTS}). The methodology is very similar to the methodology used in determining the ART values for the various components of the vessel. The only difference between the two methodologies is in the determination of the "fluence factor."

The "fluence factor" is calculated by the following expression, $f^{(0.28-0.10\log f)}$. In calculating an ART value, the fluence is determined at the 1/4 and 3/4 thickness location and is for a time period less than the license expiration date. In determining RT_{PTS}, the fluence is at the inside surface of the vessel and is for the expiration date of the license.

This is the only difference in the two methodologies. The initial RT_{NDT}, margin, and chemistry factor terms remain the same in both cases. Table 3.4.4-1 provides the values for the initial RT_{NDT}, margin, and chemistry factor terms for each of the components of the vessel.

There is one other weld in this region. This is the upper to intermediate shell, 8-203 (Heat 10137). This weld is not included in this evaluation since the fluence to this weld is lower than any of the other components (Reference 13).

TABLE 3.4.4-1
IRT_{NDT}, Margin, and Chemistry Factor Terms for Each Reactor Vessel Beltline Component

Component	Identification Number	IRT_{NDT} (Reference 12)	Margin (Reference 12)	Chemistry Factor
Intermediate Shell Longitudinal Welds	2-203-A, B, C	- 56	66	34
Lower Shell Longitudinal Welds	3-203-A, B, C	- 56	66	34
Intermediate to Lower Shell Girth Weld	9-203	- 10	56	34.1
Intermediate Shell Plate	C8009-1	- 26	34	63.6
Intermediate Shell Plate	C8009-2	0	34	54.5
Intermediate Shell Plate	C8009-3	0	34	62.2
Lower Shell Plate	C8010-1	12	34	54.5
Lower Shell Plate	C8010-2	- 28	34	53.1
Lower Shell Plate	C8010-3	- 30	34	51.0

The inside surface fluence at 32 EFPY was determined by extending the methodology used in Reference 18 to 32 EFPY. In doing so the inside surface fluence (E > 1.0 MeV) was calculated to be 4.21 E + 19 n/cm^2. Substituting the value 4.21 in the expression to determine the "fluence factor", the result is 1.37. This value is then multiplied with the chemistry factor to determine the shift in RT_{NDT} for each component. Table 3.4.4-2 lists the shift in RT_{NDT} and the RT_{PTS} for each component.

TABLE 3.4.4-2
Shift Term and RT_{PTS} Value for Each Reactor Vessel Beltline Component

Component	Identification Number	ΔRT_{NDT}	RT_{PTS}
Intermediate Shell Longitudinal Welds	2-203-A, B, C	46.58	56.58
Lower Shell Longitudinal Welds	3-203-A, B, C	46.58	56.58
Intermediate to Lower Shell Girth Weld	9-203	46.72	92.72
Intermediate Shell Plate	C8009-1	87.13	95.13
Intermediate Shell Plate	C8009-2	74.67	108.67
Intermediate Shell Plate	C8009-3	85.21	119.21
Lower Shell Plate	C8010-1	74.67	120.67
Lower Shell Plate	C8010-2	72.75	78.75
Lower Shell Plate	C8010-3	69.87	73.87

Table 3.4.4-3 compares these new values for RT_{PTS} with the values previously reported to the NRC and the screening criteria listed in 10CFR50.61.

TABLE 3.4.4-3
Comparison of New Values for RT_{PTS} to the Previous Values and the Screening Criteria

Component	Identification Number	New Values For RT_{PTS}	Previous Values for RT_{PTS} (Reference 8)	Screening Criteria
Intermediate Shell Longitudinal Welds	2-203-A, B, C	56.58	76	270
Lower Shell Longitudinal Welds	3-203-A, B, C	56.58	76	270
Intermediate to Lower Shell Girth Weld	9-203	92.72	96	300
Intermediate Shell Plate	C8009-1	95.13	126	270
Intermediate Shell Plate	C8009-2	108.67	106	270
Intermediate Shell Plate	C8009-3	119.21	106	270
Lower Shell Plate	C8010-1	120.67	118	270
Lower Shell Plate	C8010-2	78.75	68	270
Lower Shell Plate	C8010-3	73.87	66	270

As can be seen in Table 3.4.4-3, the changes in the material properties and the fluence estimates did not significantly impact the values for RT_{PTS} that were previously submitted to the Staff. In addition, Table 3.4.4-3 demonstrates adequate margin to the screening criteria listed in 10CFR50.61.

3.4.5 Decrease in Upper Shelf Energy

Position 1.2 of Reference 11 states the Charpy upper-shelf energy (USE) should be assumed to decrease as a function of fluence and copper content as indicated in Figure 2 of Reference 11. Based on the new fluence estimates and the new copper content of the beltline plates and welds, it can be seen there is a small impact on the amount of decrease in the USE. This information does not impact the unirradiated USE values.

4.0 Conclusions

As a result of the CEOG's efforts to assemble and evaluate the design and fabrication records of the ANO-2 reactor vessel beltline plates and welds, the best-estimate copper and nickel contents of these components have been revised. The impact of this revision on the ANO-2 vessel integrity evaluations has been assessed. The results of these assessments demonstrate:

- The limiting component has changed from plate C8009-1 to C8010-1. The welds in the beltline region are not limiting.
- The vessel remained operable in the past.
- The current technical specification P/T limits remain valid as they are, and therefore, the current LTOP limits remain valid.
- The period of applicability for the current technical specification P/T limits remains at 21 EFPY.
- The RT_{PTS} values for the beltline components remain significantly less than the screening criteria presented in 10CFR50.61.
- There is a small impact on the amount of decrease in the USE of the beltline welds and plates.

Based on the information presented above, the requirements of Reference 1 are considered to be fulfilled for ANO-2.