Central File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 29, 1994

MEMORANDUM FOR: All Region II Project Directors

THRU: Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

FROM: Daniel G. McDonald, Senior Project Manager
Lead Project Manager - MPA B-120
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

SUBJECT: GENERIC LETTER 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY" (MPA B-120)

The subject Generic Letter (GL) is part of the staff's program to evaluate reactor vessel structural integrity. The Materials and Chemical Engineering Branch (EMCB) has completed its review of the licensees' responses to the GL.

Enclosure 1 provides eight inserts to be sent to licensees. Insert 1 is to be sent to plants with no open issues. Insert 2 is to be sent to plants which need to confirm the applicability of Topical Reports BAW-2178P and BAW-2192P. Insert 3 is to be sent to plants to confirm the applicability of Topical Reports BAW-2178P and BAW-2192P and to confirm that the end-of-life (EOL) upper shelf energy (USE) for certain beltline materials is greater than 50 ft-lbs. Insert 4 is for St. Lucie 1 to confirm that the end-of-life (EOL) upper shelf energy (USE) for weld 2-203 is greater than 50 ft-lbs. Insert 5 is for North Anna 2 to confirm the nickel content provided for weld 05B. Insert 6 is for Robinson 2 to request that the licensee submit a request for NRC staff review and approval of the equivalent margins analysis in accordance with 10 CFR Part 50, Appendix G. Insert 7 is for Farley 1 and 2 to request that the licensee to provide a determination of the best-estimate amount of nickel

Contacts:

D. McDonald, LPM, 504-1408
E. Hackett, EMCB, 504-2751

for certain beltline welds and to justify the use of surveillance data in establishing the unirradiated upper shelf energies for certain beltline welds. Insert 8 is for Watts Bar 1 to request that the licensee acknowledge that the staff is in receipt of an equivalent margins analysis and is in the process of completing its' review.

A sample closeout letter to be sent to the PWR licensees is provided as Enclosure 2. Please replace the blank paragraph identified as "INSERT" in the sample closeout letter with the paragraph under one of the groups to which the plant belongs. Each closeout letter will include (as enclosures) two tables of data including a key to the nomenclature used. This data was compiled based on each licensee's response(s) to GL 92-01 and other information that has been previously docketed. These tables, along with a key to the nomenclature used in the tables, are provided as Enclosure 3. The closeout letter notes that the information in the tables will be used for future NRC assessments of the licensee's reactor pressure vessel(s) unless we are informed within 30 days from the date of the letter of any discrepancies.

All PWR Project Managers are requested to use the date of the closeout letter as the licensing action complete date. Please enter the date in WISP. I will provide guidance on the implementation complete date shortly after the 30-day responses are received and EMCB determines that they are acceptable.

As noted in the letters, there will be some plant-specific followup effort which is the result of (but not part of) the GL 92-01 effort. I request that the Project Managers provide me the TAC numbers they open for these plant-specific licensing actions. I will assist EMCB in tracking the closure of these actions. EMCB will issue a NUREG detailing the GL 92-01 effort and the status of all issues, including plant-specific actions, relating to reactor vessel structural integrity. The NRC considers all of the material contained in the attached files to be public information. References considered proprietary by the licensee will be so designated.

Please contact me if you have any questions. This memorandum with its enclosures is on the LAN (Filename: S:\9201REG.II) with the exception of the tables. Each PM is to use the appropriate tables for his/her plant(s) which are provided with their Project Director's copy of this memorandum.

Daniel G. McDonald, Senior Project Manager
Lead Project Manager - MPA B-120
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. List of Plants
2. Closeout Letter
3. Nomenclature and Tables

cc w/enclosures 1 and 2:
ADPR/NRR
S. Varga
J. Roe
D. Crutfchfield
DRPE/DRPW Assistant Directors
Region II Project Managers
Technical Assistant, DRPE
Technical Assistant, DRPW

ENCLOSURE 1

GENERIC LETTER (GL) 92-01 REVIEW STATUS

GROUP 1: PLANTS WITH NO OPEN ISSUES

Plant	TAC Number
Catawba 1	M83448
Catawba 2	M83449
Crystal River 3*	M83731
McGuire 1	M83480
McGuire 2	M83481
Oconee 1*	M83734
Oconee 2*	M83735
Oconee 3*	M83736
Sequoyah 1	M83513
Sequoyah 2	M83514
Shearon Harris	M83468
St. Lucie 2	M83506
Summer	M83517
Turkey Point 3*	M83742
Turkey Point 4*	M83743
Vogtle 1	M83522
Vogtle 2	M83523
Watts Bar 2	M83526

INSERT 1 (GROUP 1)
(FOR PLANTS WITHOUT THE SYMBOL "*")

We request that you verify that the information that you have provided for your facility(ies) has been accurately entered in the summary data files. No response is necessary unless an inconsistency is identified. If no comments are received within 30 days from the date of this letter, the staff will consider your actions related to GL 92-01, Revision 1, to be complete and the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel.

INSERT 2* (GROUP 1)
(FOR PLANTS WITH THE SYMBOL "*")

We request that, within 30 days, you provide confirmation of the plant-specific applicability of the Topical Reports BAW-2178P and BAW-2192P and submit a request for approval of the topical reports as the basis for demonstrating compliance with 10 CFR Part 50, Appendix G, Paragraph IV.A.1. To demonstrate that the topical reports are applicable to [your plant(s)], you must compare the limiting material properties of the [plant(s)] reactor vessel(s) to the values reported in the topical reports. This review will be a plant-specific licensing action. We further request that you verify that the information you have provided for your facility has been accurately entered in the summary data file. If no comments are made in your response to the last request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel. Once your confirmation of the applicability of the topical reports and request for approval are received, the staff will consider your actions related to GL 92-01, Revision 1, to be complete.

GROUP 2: PLANTS REQUIRING ADDITIONAL DATA TO CONFIRM THAT THE USE AT EOL IS GREATER THAN 50 FT-LB

Plant	TAC Number
North Anna 1*	M83488
St. Lucie 1*	M83505
Surry 1	M83739
Surry 2	M83740

INSERT 3 (GROUP 2)
(FOR PLANTS WITHOUT THE SYMBOL "*")

We request that, within 30 days, you provide confirmation of the plant-specific applicability of the Topical Reports; BAW-2178P and BAW-2192P and submit a request for approval of the topical reports as a basis for demonstrating compliance with 10 CFR Part 50, Appendix G, Paragraph IV.A.1. To demonstrate that the topical reports are applicable to Surry, Units 1 and 2, you must compare the limiting material properties of your reactor vessel to the values reported in the topical reports.

In addition, we have determined that additional data is required to confirm that the USE at end-of-life (EOL) for the nozzle belt to intermediate shell circumferential weld in Surry, Units 1 and 2, is greater than 50 ft-lb because you have provided a generic mean value for the unirradiated USE. These types of values are unacceptable because they do not consider material variability. When the unirradiated USE for a particular material has not been determined, you can set the USE equal to the lower tolerance limit calculated for the group of similar materials. The unirradiated USE should be determined such that there exists 95% confidence that at least 95% of the population is greater than the lower tolerance limit. If the lower tolerance limit results in a projected USE at EOL of less than 50 ft-lb, then [the licensee] must demonstrate, in accordance with Appendix G, 10 CFR Part 50, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

We request that you submit within 30 days a schedule for performing these analyses. Further, we request that you verify that the information you have provided for your facility has been accurately entered in the summary file. If no comments are made in your response to this request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel. Once your response is received and your schedule is determined to be satisfactory, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. When your analyses are submitted, they will be reviewed as a plant-specific licensing action.

<u>INSERT 4 (GROUP 2)</u>
(FOR PLANTS WITH THE SYMBOL "*")

We have determined that additional data is required to confirm that the USE at end-of-life (EOL) for one of your beltline materials, [(weld 2-203, (for St. Lucie 1)] [forging 05, (for North Anna 1)] is greater than 50 ft-lb because you have provided a generic mean value for the unirradiated USE. These types of values are unacceptable because they do not consider material variability. When the unirradiated USE for a particular material has not been determined, you can set the USE equal to the lower tolerance limit calculated for the group of similar materials. The unirradiated USE should be determined such that there exists 95% confidence that at least 95% of the population is greater than the lower tolerance limit. If the lower tolerance limit results in a projected USE at EOL of less than 50 ft-lb, than [the licensee] must demonstrate, in accordance with Appendix G, 10 CFR Part 50, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

We request that you submit within 30 days a schedule for performing these analyses. Further, we request that you verify that the information you have provided for your facility has been accurately entered in the summary file. If no comments are made in your response to this request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel. Once your response is received and your schedule is determined to be satisfactory, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. When your analyses are submitted, they will be reviewed as a plant-specific licensing action.

NORTH ANNA 2 - TAC NUMBER M83489

INSERT 5

We have determined that additional data is required to confirm the value provided for the nickel content of weld 05B of the North Anna, Unit 2, reactor vessel. The value of 0.10 provided in the GL 92-01 submittal was cited as an "estimated" value. However, the supporting data and methodology for determining the estimated value were not provided. The Pressurized Thermal Shock (PTS) Rule, 10 CFR 50.61, requires that the amounts of copper and nickel be best-estimate values. According to the PTS Rule, a mean value is acceptable for welds fabricated using the same heat number as that which matches the critical reactor vessel weld. If these values are unavailable, upper limiting values given in the material specifications to which the reactor vessel was built may be used. If not available, conservative estimates (mean plus one standard deviation) based on generic data (data from reactor vessels fabricated to the same material specification in the same shop as your vessel and in the same time period) may be used if justification is provided. If none of these alternatives are available, 1.0 percent nickel must be assumed. We request, within 30 days, that you provide the Westinghouse Owners Group (WOG) data that was used to determine the amount of nickel and that you determine the best-estimate amount of nickel in accordance with the PTS Rule, 10 CFR 50.61.

Further, we request that you verify that the information you have provided for your facility has been accurately entered in the summary file. If no comments are made in your response to this request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel. Once your response is received and your schedule is determined to be satisfactory, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. When your analyses are submitted, they will be reviewed as a plant-specific licensing action.

ROBINSON 2 - TAC NUMBER M83504

INSERT 6

The applicability of the Westinghouse Owners Group (WOG) Equivalent Margins Analysis (WCAP-13587, Rev. 1) to the Robinson, Unit 2, reactor vessel beltline materials was addressed in a letter to NRC dated November 29, 1993. Additional revised plant-specific calculations performed by Westinghouse and pertaining to the equivalent margins analysis were provided in a letter to NRC dated December 21, 1993. In these letters you did not request NRC review and approval. In addition, WCAP-13587, Rev. 1, was submitted for information only and not as a topical report. We therefore request, in accordance with the requirements of 10 CFR Part 50, Appendix G, that you submit a request for NRC review and approval of the equivalent margins analyses performed for the Robinson Unit 2, beltline materials. This can be accomplished by either requesting review and approval of the letters previously submitted and referencing WCAP-13587, Rev. 1, or by providing a plant-specific analysis independent of the WCAP-13586, Rev. 1, analysis for our review and approval.

We request that you submit, within 30 days, a schedule for completing this action. Further, we request that you verify that the information you have provided for your facility has been accurately entered in the summary file. If no comments are made in your response to this request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel. Once your response is received and your schedule is determined to be satisfactory, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. When your request is received, your analysis will be reviewed as a plant-specific licensing action.

FARLEY 1 AND 2 - TAC NUMBERS - M83461 AND M83462

INSERT 7

As a result of our Generic Letter (GL) 92-01 review, the staff has identified two open issues for each of your plants:

(1) The amounts of nickel in welds fabricated using weld wire heat numbers 33A277, 6329637 and 90099 in Farley, Unit 1, and weld wire heat numbers 5P5622 and 83640 in Farley, Unit 2, were determined as mean values from the Westinghouse Owners Group (WOG) data base. The Presurized Thermal Shock (PTS) Rule, 10 CFR 50.61, requires the amounts of copper and nickel to be best-estimate values. According to the PTS Rule, a mean value is acceptable for welds fabricated using the same heat number as that which matches the critical vessel weld. If these values are unavailable, upper limiting values given in the material specifications to which the vessel was built may be used. If not available, conservative estimates (mean plus one standard deviation) based on generic data (data from reactor vessels fabricated to the same material specification in the same shop as your vessel and in the same time period) may be used if justification is provided. If none of these alternatives are available, 1.0 percent nickel must be assumed. We request, within 30 days, that you provide the WOG data that was used to determine the amount of nickel and that you determine the best-estimate amount of nickel in accordance with the PTS Rule, 10 CFR 50.61.

(2) Surveillance data was used to determine the unirradiated upper shelf, energy values for weld wire heat numbers 6329637 and 90099 in Farley, Unit 1, and weld wire heat numbers 5P5622, 83640 and HODA in Farley, Unit 2. However, the surveillance weld data in these cases were from a different heat than the beltline welds. Additional information is required to justify use of the surveillance data. Since the surveillance data are from a different heat, a statistical analysis addressing heat variability may be appropriate. When the unirradiated USE for a particular heat of material has not been determined, you can set the USE equal to the lower tolerance limit calculated for the group of similar materials. The unirradiated USE should be determined such that there exists 95% confidence that at least 95% of the population is greater than the lower tolerance limit. If the lower tolerance limit results in a projected USE at EOL of less than 50 ft-lb, than the licensee must demonstrate, in accordance with Appendix G, 10 CFR Part 50, that lower values of USE will provide margins of safety against fracture equivalent to those required by Appendix G of Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code.

Further, we request that you verify that the information you have provided for your facilities has been accurately entered in the summary data file. If no comments are made in your response to the last request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel. Once your response is received, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. Your response to the chemical composition and surveillance weld concerns will be reviewed as plant-specific licensing actions.

WATTS BAR 1 - TAC NUMBER M83525

INSERT 8

By letter dated October 15, 1993 NRC received an Analysis entitled, "Watts Bar Unit 1 Low Upper Shelf Energy Evaluation" for review. This analysis was performed by Westinghouse and follows the approach used in the Westinghouse Owners Gorup (WOG) report on equivalent margins analyses (WCAP-13587, Rev. 1). The staff is in the process of completing its review of this analysis. The issue of the acceptability of the analysis remains open until the staff has completed its review. The staff does not require any additional information to complete the review.

We request that, within 30 days, you verify that the information you have provided for your facility has been accurately entered in the summary data file. If no comments are made in your response to the last request, the staff will use the information in the tables for future NRC assessments of your reactor pressure vessel. Once your response is received, the staff will consider your actions related to GL 92-01, Revision 1, to be complete. The equivalent margins analysis is being reviewed as a plant-specific licensing action.

SAMPLE LETTER FOR CLOSEOUT OF GL 92-01
[DATE]

Docket No.(s)

[LICENSEE ADDRESSEE]

Dear [NAME]:

SUBJECT: GENERIC LETTER (GL) 92-01, REVISION 1, "REACTOR VESSEL STRUCTURAL INTEGRITY," [UTILITY], [UNITS] (TAC NO.(s))

By letter(s) dated [ ], [utility] provided its response to GL 92-01, Revision 1. The NRC staff has completed its review of your response(s). Based on its review, the staff has determined that [utility] has provided the information requested in GL 92-01.

The GL is part of the staff's program to evaluate reactor vessel integrity for Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). The information provided in response to GL 92-01, including previously docketed information, is being used to confirm that licensees satisfy the requirements and commitments necessary to ensure reactor vessel integrity for their facilities.

A substantial amount of information was provided in response to GL 92-01, Revision 1. These data have been entered into a computerized data base designated the Reactor Vessel Integrity Database (RVID). The RVID contains the following tables: A pressurized thermal shock (PTS) table for PWRs, a pressure-temperature limits table for BWRs and an upper-shelf energy (USE) table for PWRs and BWRs. Enclosure 1 provides the PTS table(s), Enclosure 2 provides the USE table(s) for your facility(ies), and Enclosure 3 provides a key for the nomenclature used in the tables. The tables include the data necessary to perform USE and RT_{pts} evaluations. These data were taken from your response(s) to GL 92-01 and previously docketed information. References to the specific source of the data are provided in the tables.

[INSERT]

The information requested by this letter is within the scope of the overall burden estimated in GL 92-01, Revision 1, "Reactor Vessel Structural Integrity, 10 CFR 50.54(f)." The estimated average number of burden hours is 200 person hours for each addressee's response. This estimate pertains only to the identified response-related matters and does not include the time required to implement actions required by the regulations. This action is covered by the Office of Management and Budget Clearance Number 3150-0011, which expires June 30, 1994.

Sincerely,

Project Manager
PD/Branch Division
Office of Nuclear Reactor Regulation

Enclosures:
1. Pressurized Thermal Shock or Pressure-Temperature Limit Table(s)
2. Upper-Shelf Energy Table(s)
3. Nomenclature Key

cc w/enclosures:
See next page

Enclosure 3

Nomenclature and Tables

PRESSURIZED THERMAL SHOCK AND USE TABLES FOR ALL PWR PLANTS

NOMENCLATURE

Pressurized Thermal Shock Table

Column 1: Plant name and date of expiration of license.
Column 2: Beltline material location identification.
Column 3: Beltline material heat number; for some welds that a single-wire or tandem-wire process has been reported, (S) indicates single wire was used in the SAW process, (T) indicates tandem wire was used in the SAW process.
Column 4: End-of-life (EOL) neutron fluence at vessel inner wall; cited directly from inner diameter (ID) value or calculated by using Regulatory Guide (RG) 1.99, Revision 2 neutron fluence attenuation methodology from the quarter thickness (T/4) value reported in the latest submittal (GL 92-01, PTS, or P/T limits submittals).
Column 5: Unirradiated reference temperature.
Column 6: Method of determining unirradiated reference temperature (IRT).

Plant-Specific
This indicates that the IRT was determined from tests on material removed from the same heat of the beltline material.

MTEB 5-2
This indicates that the unirradiated reference temperature was determined from following MTEB 5-2 guidelines for cases where the IRT was not determined using American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, NB-2331, methodology.

Generic
This indicates that the unirradiated reference temperature was determined from the mean value of tests on material of similar types.

Column 7: Chemistry factor for irradiated reference temperature evaluation.
Column 8: Method of determining chemistry factor

Table
This indicates that the chemistry factor was determined from the chemistry factor tables in RG 1.99, Revision 2.

Calculated
This indicates that the chemistry factor was determined from surveillance data via procedures described in RG 1.99, Revision 2.

Column 9: Copper content; cited directly from licensee value except when more than one value was reported. (Staff used the average value in the latter case.)

No Data
This indicates that no copper data has been reported and the default value in RG 1.99, Revision 2, will be used by the staff.

Column 10: Nickel content; cited directly from licensee value except when more than one value was reported. (Staff used the average value in the latter case.)

No Data

This indicates that no nickel data has been reported and the default value in RG 1.99, Revision 2, will be used by the staff.

Upper Shelf Energy Table

Column 1: Plant name and date of expiration of license.
Column 2: Beltline material location identification.
Column 3: Beltline material heat number; for some welds that a single-wire or tandem-wire process has been reported, (S) indicates single wire was used in the SAW process. (T) indicates tandem wire was used in the SAW process.
Column 4: Material type; plate types include A 533B-1, A 302B, A 302B Mod., and forging A 508-2; weld types include SAW welds using Linde 80, 0091, 124, 1092, ARCOS-B5 flux, Rotterdam welds using Graw Lo, SMIT 89, LW 320, and SAF 89 flux, and SMAW welds using no flux.
Column 5: EOL upper-shelf energy (USE) at T/4; calculated by using the EOL fluence and either the cooper value or the surveillance data. (Both methods are described in RG 1.99, Revision 2.)

EMA
This indicates that the USE issue may be covered by either owners group or plant-specific equivalent margins analyses.

Column 6: EOL neutron fluence at T/4 from vessel inner wall; cited directly from T/4 value or calculated by using RG 1.99, Revision 2 neutron fluence attenuation methodology from the ID value reported in the latest submittal (GL 92-01, PTS, or P/T limits submittals).

Column 7: Unirradiated USE.

<u>EMA</u>
This indicates that the USE issue may be covered by either owners group or plant-specific equivalent margins analyses.

Column 8: Method of determining unirradiated USE

<u>Direct</u>
For plates, this indicates that the unirradiated USE was from a transverse specimen. For welds, this indicates that the unirradiated USE was from test date.

<u>65%</u>
This indicates that the unirradiated USE was 65% of the USE from a longitudinal specimen.

<u>Generic</u>
This indicates that the unirradiated USE was reported by the licensee from other plants with similar materials to the beltline material.

<u>NRC generic</u>
This indicates that the unirradiated USE was derived by the staff from other plants with similar materials to the beltline material.

<u>10, 30, 40, or 50 °F</u>
This indicates that the unirradiated USE was derived from Charpy test conducted at 10, 30, 40, or 50 °F.

<u>Surv. Weld</u>
This indicates that the unirradiated USE was from the surveillance weld having the same weld wire heat number.

<u>Equiv. to Surv. Weld</u>
This indicates that the unirradiated USE was from the surveillance weld having different weld wire heat number.

<u>Sister Plant</u>
This indicates that the unirradiated USE was derived by using the reported value from other plants with the same weld wire heat number.

<u>Blank</u>
indicates that there is insufficient data to determine the unirradiated USE.

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Catawba 1	Int. shell Forging 05	411343	2.52E19	-8°F	Plant Specific	58	Table	0.09	0.86
EOL: 12/06/ 2024	Lower shell Forging 04	527708	2.52E19	-13°F	Plant Specific	26	Table	0.04	0.83
	Circ. Weld W05	895075	2.52E19	-51°F	Plant Specific	68	Table	0.05	0.74

References for Catawba 1

Chemical composition and IRT_{ndt} data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity

The chemical composition for Intermediate Shell Forging 05 is an average of data found in the GL 92-01 response and WCAPs 9734 and 11527.

Fluence data are from Table 6-18 of WCAP-13720. The peak fluence was used.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
Catawba 1 EOL: 12/06/ 2024	Int. shell Forging 05	411343	A 508-2	106	1.52E19	134	Direct
	Lower shell Forging 04	527708	A 508-2	106	1.52E19	134	Direct
	Circ. Weld W05	895075	Grau Lo, SAW	101	1.52E19	128	Direct

References

The unirradiated USE values for the Forgings and the Circ. Weld are from Table C-1 of WCAP-13720.

The EOL USE values for the forgings were conservatively calculated using Fig. 2 of RG 1.99, Revision 2 assuming 0.10% Cu.

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Catawba 2	Int. shell	B8605-1	3.05E19	15°F	Plant Specific	51	Table	0.08	0.62
EOL: 2/24/2026	Int. shell	B8605-2	3.05E19	33°F	Plant Specific	44	Table	0.07	0.61
	Int. shell	B8616-1	3.05E19	12°F	Plant Specific	31	Table	0.05	0.59
	Lower shell	B8806-1	3.05E19	6°F	Plant Specific	31	Table	0.05	0.56
	Lower shell	B8806-2	3.05E19	-10°F	Plant Specific	31	Table	0.05	0.59
	Lower shell	B8806-3	3.05E19	8°F	Plant Specific	31	Table	0.05	0.59
	Axial & Girth Welds G1.45	Linde 0091 Flux 3536	3.05E19	-80°F	Plant Specific	38	Table	0.04	0.15

References for Catawba 2

Chemical composition and IRT_{ndt} data are from Table A-1 of WCAP-11941. Chemical composition for B8605-1 and weld G1.45 are the average value from WCAP-10868 and WCAP-11941.

Fluence data are from Table 6-13 of WCAP-11941.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
Catawba 2 EOL: 2/24/2026	Int. shell B8605-1		A 533B-1	69	1.835E19	89	Direct
	Int. shell B8605-2		A 533B-1	64	1.835E19	82	Direct
	Int. shell B8616-1		A 533B-1	72	1.835E19	92	Direct
	Lower shell B8806-1		A 533B-1	65	1.835E19	83	Direct
	Lower shell B8806-2		A 533B-1	80	1.835E19	102	Direct
	Lower shell B8806-3		A 533B-1	82	1.835E19	105	Direct
	Welds G1.45	83648	Linde 0091, SAW	101	1.835E19	130	Direct

<u>References</u>

Chemical composition, UUSE, fluence and IRTndt data are from Table A-1 of WCAP-11941. Fluence data are from Table 6-13 of WCAP-11941.

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Crystal River 3 EOL: 12/3/2016	Nozzle Belt Forging	AZJ94	7.53E18	3°F	Generic	66.2	Table	0.10	0.72
	Upper Shell	C4344-1	8.56E18	20°F	Plant Specific	119.64	Calculated	0.20	0.54
	Upper Shell	C4344-2	8.56E18	20°F	Plant Specific	141.8	Table	0.20	0.54
	Lower Shell	C4347-1	8.22E18	-10°F	Plant Specific	82.6	Table	0.12	0.58
	Lower Shell	C4347-2	8.22E18	45°F	Plant Specific	82.6	Table	0.12	0.58
	Upper shell axial weld WF-8/WF-18	8T1762	7.96E18	-5°F	Generic	152.25	Table	0.20	0.55
	Upper shell circ. weld SA-1769 (ID 40%)	71249	7.53E18	-5°F	Generic	174.65	Table	0.26	0.61
	Upper/lower shell circ. weld WF-70	72105	8.22E18	-26°F	NRC Generic	148.02	Calculated	0.35	0.59
	Lower shell axial weld SA-1580	8T1762	6.98E18	-5°F	Generic	152.25	Table	0.20	0.55

References

IRT_{ndt} for WF-70, Upper/Lower shell circ. weld determined by a method that was reviewed by the staff in a February 22, 1994 letter from C.Y. Shiraki (USNRC) to D.L. Farrar (Comm. Ed. Co.). In order to utilize this value of IRT_{ndt}, the licensee will need an exemption from determining the IRT_{ndt} in accordance with 10CFR 50.61(b)(2)(i).

Chemical composition, fluence data, and IRT_{ndt} data of all materials except for Nozzle Belt Forging, are from BAW-2166.

Chemical composition of Nozzle Belt. Forging is from BAW-2049.

Chemistry factor for Upper/Lower circ. weld WF-70, was calculated from surveillance welds irradiated in Oconee 2, Oconee 3, and Davis-Besse capsules. The welds in these capsules were fabricated using the same weld were heat number as WF-70.

The IRT_{ndt} of the Nozzle Belt Forging is the mean value for data reported in BAW 10046P. The standard deviation for this data is 31°F.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
Crystal River 3 EOL: 12/3/2016	Nozzle Belt Forging	AZJ94	A 508-2	55	4.52E18	65	Generic 10°F/65%
	Upper Shell	C4344-1	A 533B-1	66	5.14E18	88	Direct
	Upper Shell	C4344-2	A 533B-1	66	5.14E18	88	Direct
	Lower Shell	C4347-1	A 533B-1	98	4.94E18	119	Direct
	Lower Shell	C4347-2	A 533B-1	71	4.94E18	86	Direct
	Upper shell axial weld WF-8/WF-18	8T1762	Linde 80, SAW	EMA[2]	4.78E18	EMA[2]	Generic
	Upper shell circ. weld SA-1769 (ID 40%)	71249	Linde 80, SAW	EMA[2]	4.52E18	EMA[2]	Generic
	Upper/ lower shell circ. weld WF-70	72105	Linde 80, SAW	EMA[2]	4.94E18	EMA[2]	Generic
	Lower shell axial weld SA-1580	8T1762	Linde 80, SAW	EMA[2]	4.19E18	EMA[2]	Generic

References

Chemical composition and fluence data are from BAW-2166.

UUSE for upper shell and lower shell plates per BAW 1820, TL orientation

UUSE for AZJ94 calculated from 10°F data reported in BAW-2166 (65% Correction Factor).

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[2]Licensee must confirm applicability of Topical Reports BAW-2178P and BAW-2192P

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL/EFPY	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Farley 1 EOL: 6/25/2017	Int. Shell B6903-2	C6294-1	3.75E19	0°F	MTEB 5-2	91	Table	0.13	0.60
	Int. Shell B6903-3	C6308-2	3.75E19	10°F	MTEB 5-2	82.2	Table	0.12	0.56
	Lower Shell B6919-1	C6940-1	3.75E19	15°F	MTEB 5-2	88.831	Calculated	0.14	0.55
	Lower Shell B6919-2	C6897-2	3.75E19	5°F	MTEB 5-2	98.2	Table	0.14	0.56
	Int. Shell Axial Welds	33A277	1.24E19	-56°F	Generic	78.689	Calculated	0.25	0.21 [9]
	Circ. Weld	6329637	3.75E19	-56°F	Generic	113	Table	0.23	0.20 [9]
	Lower Shell Axial Welds	90099	1.24E19	-56°F	Generic	92	Table	0.17	0.20 [9]

REFERENCES FOR FARLEY 1:

Fluence, IRT_{ndt} and chemistry values from November 23, 1993, letter from D. Morey (SNOC) to USNRC Document Control Desk, subject: Responses to Requests for Additional Information Regarding GL 92-01.

[9] Chemical composition from mean value of WOG data. Additional information required.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL/EFPY	1/4T Neutron Fluence at EOL/EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Farley 1 EOL: 6/25/2017	Int. Shell B6903-2	C6294-1	A 533B-1	73	2.34E19	99	65%
	Int. Shell B6903-3	C6308-2	A 533B-1	65	2.34E19	87	65%
	Lower Shell B6919-1	C6940-1	A 533B-1	66	2.34E19	86	65%
	Lower Shell B6919-2	C6897-2	A 533B-1	62	2.34E19	86	65%
	Int. Shell Axial Welds	33A277	Linde 1092, SAW	115	7.73E18	149	Surv. Weld
	Circ. Weld	6329637	Linde 0091, SAW	115 [10]	2.34E19	149 [10]	Equiv. to Surv. Weld
	Lower Shell Axial Welds	90099	Linde 0091, SAW	115 [10]	7.73E18	149 [10]	Equiv. to Surv. Weld

<u>References for Farley 1</u>

Fluence, heat number and UUSE values from November 23, 1993, letter from D. Morey (SNOC) to USNRC Document Control Desk, subject: Responses to Requests for Additional Information Regarding GL 92-01.

[10]Surveillance weld is from a different heat than beltline welds, additional information required to confirm licensee's value

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL/EFPY	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Farley 2 EOL: 3/31/2021	Int. Shell B7203-1	C6309-2	3.8E19	15°F	Plant Specific	100	Table	0.14	0.60
	Int. Shell B7212-1	C7466-1	3.8E19	-10°F	Plant Specific	145.0	Calculated	0.20	0.60
	Lower Shell B7210-1	C6888-2	3.8E19	18°F	Plant Specific	89.8	Table	0.13	0.56
	Lower Shell B7210-2	C6293-1	3.8E19	10°F	Plant Specific	98.7	Table	0.14	0.57
	Circ. Weld 11-923	5P5622	3.8E19	-40°F	Plant Specific	76	Table	0.13	0.20 [9]
	Lower Shell Axial Welds 20-923 A/B	83640	1.23E19	-70°F	Plant Specific	49	Table	0.05	0.20 [9]
	Int. Shell Axial Welds 19-923A	HODA SMAW	1.23E19	-56°F	Generic	10.01	Calculated	0.02	0.96
	Int. Shell Axial Welds 19-923B	BOLA SMAW	1.23E19	-60°F	Plant Specific	10.01	Calculated	0.02	0.93

REFERENCES FOR FARLEY 2:

Fluence, IRT_{ndt} and chemistry values from November 23, 1993, letter from D. Morey (SNOC) to USNRC Document Control Desk, subject: Responses to Requests for Additional Information Regarding 92-01.

[9] Chemical composition from mean value of WOG data. Additional information required to confirm value.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL/EFPY	1/4T Neutron Fluence at EOL/EFPY	Unirrad. USE	Method of Determin. Unirrad. USE
Farley 2 EOL: 3/31/2021	Int. Shell B7203-1	C6309-2	A 533B-1	72	2.369E19	100	Direct
	Int. Shell B7212-1	C7466-1	A 533B-1	62	2.369E19	100	Direct
	Lower Shell B7210-1	C6888-2	A 533B-1	76	2.369E19	103	Direct
	Lower Shell B7210-2	C6293-1	A 533B-1	72	2.369E19	99	Direct
	Circ. Weld 11-923	5P5622	Linde 0091, SAW	110 [10]	2.369E19	148 [10]	Equiv. to Surv. Weld
	Lower Shell Axial Welds 20-923A/B	83640	Linde 0091, SAW	131 [10]	7.67E18	148 [10]	Equiv. to Surv. Weld
	Int. Shell Axial Welds 19-923A	HODA	SMAW	131 [10]	7.67E18	148 [10]	Equiv. to Surv. Weld
	Int. Shell Axial Welds 19-923B	BOLA	SMAW	131	7.67E18	148	Surv. Weld

REFERENCES FOR FARLEY 2:

Fluence, heat number and UUSE values from November 23, 1993, letter from D. Morey (SNOC) to USNRC Document Control Desk, subject: Responses to Requests for Additional Information Regarding GL 92-01.

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[10] Surveillance weld is from a different heat than beltline welds, additional information required to confirm licensee's value

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
McGuire 1	Int. shell B5012-1	C4387-2	2.02E19	34°F	Plant Specific	54.51	Calculated	0.11	0.61
EOL: 6/12/2021	Int. shell B5012-2	C4417-3	2.02E19	0°F	Plant Specific	100	Table	0.14	0.61
	Int. shell B5012-3	C4377-2	2.02E19	-13°F	Plant Specific	75	Table	0.11	0.66
	Lower shell B5013-1	C4315-1	2.02E19	0°F	Plant Specific	99.1	Table	0.14	0.58
	Lower shell B5013-2	C4374-2	2.02E19	30°F	Plant Specific	65	Table	0.10	0.51
	Lower shell B5013-3	C4371-2	2.02E19	15°F	Plant Specific	65	Table	0.10	0.55
	Int. shell axial welds M1.22	20291 and 12008	1.46E19	-50°F	Plant Specific	171.06	Calculated	0.20	0.87
	Int. to lower shell circ. weld G1.39	83640	2.02E19	-70°F	Plant Specific	39.8	Table	0.05	0.12
	3-442 ABC Lower shell axial welds M1.32	21935 and 12008	1.46E19	-56°F	Generic	209.6	Table	0.22	0.86

References for McGuire 1

IRT_{ndt} data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

Fluence data are from the March 17, 1994 letter to USNRC subject: Pressurized Thermal Shock

Chemical compositions are from the response to GL 92-01 dated 12/20/93 and from the March 17, 1994 letter on Pressurized Thermal Shock.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
McGuire 1 EOL: 6/12/2021	Int. shell B5012-1	C4387-2	A 533B-1	71	1.20E19	89	65%
	Int. shell B5012-2	C4417-3	A 533B-1	68	1.20E19	89	65%
	Int. shell B5012-3	C4377-2	A 533B-1	81	1.20E19	100	65%
	Lower shell B5013-1	C4315-1	A 533B-1	64	1.20E19	84	65%
	Lower shell B5013-2	C4374-2	A 533B-1	77	1.20E19	96	65%
	Lower shell B5013-3	C4371-2	A 533B-1	68	1.20E19	85	65%
	Int. shell axial welds M1.22	20291 and 12008	Linde 1092, SAW	70	0.87E19	110	Direct
	Int. to lower shell 9-4442 circ. weld G1.39	83640	Linde 0091, SAW	101	1.20E19	126	Direct
	Lower shell axial welds M1.32	21935 and 12008	Linde 1092, SAW	59	0.87E19	90	Direct

References

Chemical composition and UUSE data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

UUSE data for welds M1.22, G1.39, and M1.32 are from Table A-1 of WCAP-10786.

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
McGuire 2	Forging 05 Int. shell	526840	2.04E19	-4°F	Plant Specific	87.98	Calculated	0.16	0.85
EOL: 3/3/2023	Forging 04 Lower shell	411337-11	2.04E19	-30°F	Plant Specific	115.8	Table	0.15	0.88
	Int. to Lower shell Welds Shell W05	895075	2.04E19	-68°F	Plant Specific	33.48	Calculated	0.03	0.70

<u>References for McGuire 2</u>

Chemical composition and IRT_{ndt} data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

Fluence data are from Table 6-22 of WCAP-13516.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
McGuire 2	Int. shell Forging 05	526840	A 508-2	74	1.225E19	100	Direct
EOL: 3/3/2023	Lower shell Forging 04	411337-11	A 508-2	71	1.225E19	97	65%
	Int. lower shell Welds W05	895075	Grau Lo, SAW	112	1.225E19	140	Direct

References

Chemical composition and UUSE data are from July 3, 1992, letter from H. B. Tucker (DPCo) to USNRC Document Control Desk, subject: McGuire Nuclear Station, Units 1 and 2; Catawba Nuclear Station Units 1 and 2; Oconee Nuclear Station, Units 1, 2, and 3: Response to Generic Letter 92-01, Revision 1, Reactor Vessel Structural Integrity.

Fluence data are from Table 6-22 of WCAP-13516.

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
North Anna 1 EOL: 4/1/2018	Nozzle shell forging 05	990286/ 295213	2.51E18	6°F	MTEB 5-2	121.5	Table	0.16	0.74
	Int. shell forging 04	990311/ 298244	3.95E19	17°F	Plant Specific	86	Table	0.12	0.82
	Lower shell forging 03	990400/ 292332	3.95E19	38°F	Plant Specific	73.503	Calculated	0.16	0.80
	Weld 04	25531	3.95E19	19°F	Plant Specific	93.089	Calculated	0.09	0.11
	Weld 05A	25295	2.78E18	0°F	Generic	138.5	Table	0.30	0.17
	Weld 05B	4278	2.78E18	0°F	Generic	58.5	Table	0.11	0.11

References

The nickel contents for weld 05A and 05B are values from Sequoyah 1&2. (Same weld wire heat numbers).

Chemical composition and IRT_{ndt} data are from BAW-2168, which is attached to the GL 92-01 response.

Fluence data:

WCAP-11777: ID EOL fluence is 3.95E19 n/cm²

Table 2-1 of BAW-2146, which is attached to December 27, 1991, letter from W. L. Stewart (VPCo) to USNRC Document Control Desk, subject: Request to Change Technical Specifications: Pressure/Temperature Limitations, Low Temperature/Overpressure Protection System Setpoints, states that forging 05, and welds 05A and 05B have fluences that differ from forgings 03 and 04, and weld 04.

Note:

Chemical composition values for forging as are averages from the beltline material data and the surveillance data.

A margin of 69°F (σ_I = 20°F $\sigma\Delta$ = 28°F) has to be used for weld 05A and weld 05B for which a generic IRT_{NDT} of 0°F has been derived.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
North Anna 1	Nozzle shell forging 05	990286/ 295213	A 508-2	62	1.74E18	75[7]	Generic
EOL: 4/1/2018	Int. shell forging 04	990311/ 298244	A 508-2	68	2.49E19	92	Direct
	Lower shell forging 03	990400/ 292332	A 508-2	58	2.49E19	85	Direct
	Circ. Weld Weld 04	25531	SMIT 89, SAW	71	2.49E18	102	Direct
	Nozzle to Int. Shell Weld 05A	25295	SMIT 89, SAW	78	1.74E18	111	Sister Plant
	Nozzle to Int. Shell Weld 05B	4278	SMIT 89, SAW	---	1.74E18	105	Sister Plant

References

The fluence data for weld 04 is from June 29, 1992 letter to NRC (Response to GL 92-01); fluence data for other materials are from September 23, 1993 letter to NRC (Response to GL 92-01 RAI).

Chemical composition and UUSE data for forging 03 are from BAW-2168, which is attached to the GL 92-01 response.

UUSE data for forging 04 and weld 04 are from BAW-1911, Rev. 1.

Note: Weld 05A is 94% of thickness of the Nozzle to intermediate shell weld and 05B is the remainder. Therefore, it is not necessary to evaluate the EOL USE for weld 05B because it is not at the 1/4T location.

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[7] Additional information required to confirm value

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{ndt}	Method of Determin. IRT_{ndt}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
North Anna 2 EOL: 8/21/2020	Upper shell forging 05	990598/ 291396	4.47E19	9°F	MTEB 5-2	51	Table	0.08	0.77
	Int. shell forging 04	990496/ 292424	4.47E19	75°F	Plant Specific	35.112	Calculated	0.10	0.85
	Lower shell forging 03	990533/ 207355	4.47E19	56°F	Plant Specific	96	Table	0.13	0.83
	Weld 04	716126	4.47E19	-48°F	Plant Specific	10.398	Calculated	0.09	0.08
	Weld 05A	4278	4.47E19	0°F	Generic	58.5	Table	0.11	0.11
	Weld 05B	801	4.47E19	0°F	Generic	87.0	Table	0.18	0.10 [7]

References

The nickel content for weld 05A is from Sequoyah 2 (the same weld wire heat number and the same flux).

BAW-2168, which is attached to June 29, 1992, letter from W. L. Stewart (VPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Reactor Vessel Structural Integrity, contains chemical composition and the initial RT_{ndt} (IRT_{ndt}) data for all the beltline materials

Fluence is from Table 6-13 of WCAP-12497

Note:

Chemical composition values for forging 04 are averages from the beltline material data and the surveillance data.

A margin of 69°F (σ_I = 20°F $\sigma\Delta$ = 28°F) has to be used for weld 05A and weld 05B for which a generic IRT_{ndt} of 0°F has been derived.

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[7] Additional information required to confirm value

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
North Anna 2 FOL: 8/21/2020	Upper shell forging 05	990598/ 291396	A 508-2	64	2.0E18	74	Equiv. to forging 04
	Int. shell forging 04	990496/ 292424	A 508-2	51	2.82E19	74	Direct
	Lower shell forging 03	990533/ 207355	A 508-2	58	2.82E19	80	Direct
	Weld 04	716126	LW 320, SAW	69	2.82E19	107	Direct
	Weld 05A	4278	SMIT 89, SAW	86	2.82E19	105	Sister Plant
	Weld 05B	801	SMIT 89, SAW	---	2.82E19	---	---

References

BAW-2168, which is attached to June 29, 1992, letter from W. L. Stewart (VPCo) to USNRC Document Control Desk, subject: Response to Generic Letter 92-01, Reactor Vessel Structural Integrity, contains chemical composition data for all the beltline materials. However, it contains UUSEs for forging 04 and weld 04 only.

Fluence and UUSEs for forgings 05 and 03 are from December 29, 1992 letter to NRC.

Note: Weld 05A is 94% of thickness of the Nozzle to intermediate shell weld and 05B is the remainder. Therefore, it is not necessary to evaluate the EOL USE for weld 05B because it is not at the 1/4T location.

Summary File for Pressurized Thermal Shock

Plant Name	Beltline Ident.	Heat No. Ident.	ID Neut. Fluence at EOL	IRT_{NDT}	Method of Determin. IRT_{NDT}	Chemistry Factor	Method of Determin. CF	%Cu	%Ni
Oconee 1	Lower nozzle belt forging	AHR54	1.18E18	3°F	Plant Specific	119.25	Table	0.16	0.65
EOL: 2/6/2013	Upper shell	C3278-1	9.04E18	1°F	Plant Specific	83	Table	0.12	0.60
	Upper shell	C3265-1	9.04E18	1°F	Plant Specific	54.56	Calculated	0.10	0.50
	Int. shell	C2197-2	7.96E18	1°F	Plant Specific	104.5	Table	0.15	0.50
	Lower shell	C2800-1	8.68E18	1°F	Plant Specific	74.45	Table	0.11	0.63
	Lower shell	C2800-2	8.68E18	1°F	Plant Specific	74.45	Table	0.11	0.63
	Nozzle/int shell circ. weld SA-1135	61782	1.18E18	-5°F	Generic	133.94	Calculated	0.25	0.54
	Int/lower shell circ. weld SA-1229	71249	7.96E18	-5°F	Generic	181.6	Table	0.26	0.61
	Upp./lower shell circ. weld SA-1585	72445	8.68E18	-5°F	Generic	146.09	Calculated	0.21	0.59
	Int. shell axial welds SA-1073	1P0962	6.28E18	-5°F	Generic	170.6	Table	0.21	0.64
	Upper shell axial welds SA-1493	8T1762	7.23E18	-5°F	Generic	152.25	Table	0.20	0.55
	Lower shell axial welds SA-1430	8T1762	7.29E18	-5°F	Generic	152.25	Table	0.20	0.55
	Lower shell axial welds SA-1426	8T1762	7.29E18	-5°F	Generic	152.25	Table	0.20	0.55

References for Oconee 1

Fluence, chemical composition, and IRT_{NDT}s are from BAW-2166.

Chemistry Factor for SA-1585 was calculated from Point Beach and Crystal River surveillance data that was reported in BAW 1803, Rev. 1. The surveillance welds were fabricated using the same weld wire heat number as SA-1585.

Chemistry Factor for SA-1135 was calculated from Ginna and Davis-Besse surveillance data that was reported in BAW-1803, Rev. 1. The surveillance welds were fabricated with the same weld wire Heat Number as SA-1135.

Summary File for Upper Shelf Energy

Plant Name	Beltline Ident.	Heat No.	Material Type	1/4T USE at EOL	1/4T Neutron Fluence at EOL	Unirrad. USE	Method of Determin. Unirrad. USE
Oconee 1 EOL: 2/6/2013	Lower nozzle belt forging	AHR54	A 508-2	107	7.1E17	124	Direct
	Upper shell	C3278-1	A 302B	75	5.45E18	91	Direct
	Upper shell	C3265-1	A 302B	90	5.45E18	108	Direct
	Int. shell	C2197-2	A 302B	73	4.80E18	91	Direct
	Lower shell	C2800-1	A 302B	76	5.23E18	91	Direct
	Lower shell	C2800-2	A 302B	99	5.23E18	119	Direct
	Nozzle/ int. shell circ. weld SA-1135	61782	Linde 80, SAW	EMA[2]	7.1E18	EMA[2]	Generic
	Int/lower shell circ. weld ID 61% SA-1229	71249	Linde 80, SAW	EMA[2]	4.80E18	EMA[2]	Generic
	Upp./lower shell circ. weld SA-1585	72445	Linde 80, SAW	EMA[2]	5.23E18	EMA[2]	Generic
	Int. shell axial welds SA-1073	1P0962	Linde 80, SAW	EMA[2]	3.79E18	EMA[2]	Generic
	Upper shell axial welds SA-1493	8T1762	Linde 80, SAW	EMA[2]	4.36E18	EMA[2]	Generic
	Lower shell axial welds SA-1430	8T1762	Linde 80, SAW	EMA[2]	4.39E18	EMA[2]	Generic

[2] Licensee must confirm applicability of Topical Reports BAW-2178P and BAW-2192P