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December 9, 2003

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Mr. Russell Arrighi (Mail Stop O-11F1)
Office of Nuclear Reactor Regulation
Washington, D.C. 20555-0001

Subject: Open and Confirmatory Item Responses to Ginna SER with Open Items
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Arrighi:

RG&E is providing information supplementing our September 16, 2003 response to the NRC's Safety Evaluation Report with Open Items, dated October 2003. Revisions to our September 16 responses are noted where applicable.

I declare under penalty of perjury under the laws of the United States of America that I am authorized by RG&E to make this submittal and that the foregoing is true and correct.

Very truly yours,

Executed on December 9, 2003

Robert C. Mecredy

Robert C. Mecredy

Attachments

cc: Mr. Russ Arrighi, Project Manager (Mail Stop O-11F1)
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U.S. NRC Ginna Senior Resident Inspector

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List of Regulatory Commitments

The following table identifies those actions committed to by Rochester Gas & Electric (RG&E) in this document and in our September 16, 2003 submittal on the same subject. Any other statements in these submittals are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. George Wrobel, License Renewal Project Manager at (585) 771-3535.

REGULATORY COMMITMENT		DUE DATE
OI 2.3.3.2-2 (September 16, 2003)	Add CCW makeup water piping, valves and pumps from the RMWT to valve MOV 823 into scope of license renewal.	July 2004
OI 3.1.2.3.3-1 (September 16, 2003)	Submit Reactor Vessel internals program for staff review.	September 2007
OI 3.6-2 (September 16, 2003)	Perform joint resistance tests when visual inspections of PVC boots or other materials of construction indicate that the joint may be overheating.	Prior to September 2009
OI 2.3.3.3-1	Add spent fuel pool makeup path from RWST to the SFP into scope of license renewal.	July 2004
OI 2.3.3.6-1	Add fire service water booster pump and associated valves and piping back to the SW system into the scope of license renewal.	July 2004
OI 2.5-1	Add medium voltage cables M0089 and M0108 into the scope of license renewal, and develop aging management program consistent with NUREG-1801, Section XI.E3.	July 2004
OI B2.1.28-1	Withdraw surveillance capsule in Spring 2005, and test prior to the time that the fluence level of Capsule S is exceeded. Withdraw last surveillance capsule shortly after accumulating fluence equivalent to 80 years of operation.	Spring 2005; Prior to 2012 ~2011 RFO

OI B2.1.36-1	Perform inspections of thimble tubes for wear and SCC each refueling outage. VT-1 quality inspect SS fillet weld joining the BMI guide tube to the end of each BMI penetration, as well as the 82/182 weld between the SS safe end and the lower penetration nozzle, each refueling outage.	RFO's beginning in 2005: wear of thimble tubes RFO's beginning in 2009: SCC of thimble tubes RFO's beginning in 2005
CI 3.3.2.3.4-1	Submit License Amendment Request to incorporate specific particulate testing for diesel generator fuel oil, and eliminate the need for the "clear and bright" method of ASTM D4176.	December 2004

Attachment 1: Response to Open and Confirmatory Items

OI 2.3.3.2-1

The applicant did not provide an adequate basis in its response to RAI 2.3.3.2-1 dated May 23, 2003, for concluding that a failure in the out-of-scope piping will not result in failure of the component cooling water (CCW) system in performing its intended functions. The staff cannot make its finding regarding the acceptability of the applicant's basis without information such as the available methods of detecting piping failure, the inventory of CCW that could be lost through failed piping from the time of detection to failure of the component cooling water system, the rate of loss of inventory through a failed pipe considering that the system is pressurized, and the time necessary for reasonable assurance that operators could identify and isolate the failed piping.

Reply - Note that this is a revision to our September 16, 2003 response.

RG&E is adding that section of the CCW system serving the post-accident sampling system coolers, between valves 747A/K and 747B/J, as well as the piping, tubing, and valve bodies associated with FIC 649, 650, and 651, into the scope of license renewal. The rest of the CCW system not in scope consists of 3/4" or smaller tubing and associated valve bodies. If leakage were to occur, it would be on the order of a few gallons per minute. Conservatively assuming a flow rate of 10 gpm, the CCW surge tank could accommodate about 90 minutes of unmitigated leakage. This would be more than enough time for the operators to recognize lower tank level (low alarm at 900 gallons), and initiate makeup flow from the reactor makeup water tank. The CCW makeup water pumps, with a capacity of 60 gpm, can provide adequate makeup flow from this tank, which has a capacity of 75,000 gallons, until operators could isolate the leaking section of CCW piping. All of the necessary isolation valves are in locations accessible to perform post-accident operations.

All of the components being added to the scope of license renewal in the CCW system are already included within the component groups listed in table 2.3.3-2 of the application. Specifically:

Component Group	Passive Function	Aging Management Reference
CS Components	Pressure Boundary	Table 3.4-1 line number (13)
Fasteners (Bolting)	Joint Integrity	Table 3.4-1 line number (13) Table 3.4-1 line number (23) Table 3.4-2 line number (80)

Component Group	Passive Function	Aging Management Reference
Heat Exchanger	Pressure Boundary	Table 3.4-1 line number (5) Table 3.4-1 line number (14) Table 3.4-2 line number (151)
Indicator	Pressure Boundary	Table 3.4-1 line number (170) Table 3.4-1 line number (179)
Pipe	Pressure Boundary	Table 3.4-1 line number (15) Table 3.4-1 line number (14)
Valve Body	Pressure Boundary	Table 3.4-1 line number (15) Table 3.4-1 line number (14)

<u>OI 2.3.3.3-1</u>

By letter dated March 21, 2003, the staff requested that the applicant justify the exclusion of the alternate spent fuel pool (SFP) makeup water supply piping and valves from the scope of license renewal and AMR (RAI 2.3.3.3-2). By letter dated May 13, 2003, the applicant responded that Ginna was built before RG 1.13 was issued. The applicant further stated that RG 1.13 is used as guidance, but is not a requirement.

The staff cannot reconcile the applicant's argument with the fact that these alternative makeup water supply paths are relied upon in Ginna's CLB not only to offset boil-off due to the loss of SFP cooling, but also to mitigate potential leaks in the SFP liner. The 1998 staff approval of the re-racking of the Ginna SFP was based, in part, on redundancy in the SFP makeup water supply. The applicant specifically cited the refueling water storage tank (RWST) and chemical and volume control system (CVCS) holdup tanks as alternate sources of SFP makeup in an F-RAI response dated November 11, 1997. Although these makeup water paths are non safety-related, they are within the scope of 10 CFR Part 54 because their failure could prevent satisfactory performance of functions necessary to prevent or mitigate significant offsite exposures resulting from SFP accidents. The Statements of Consideration for 10 CFR Part 54 state that "the Commission believes it inappropriate to permit generic exclusion of redundant, long-lived, passive structures and components." In other words, redundancy is not an adequate basis in itself to exclude a system from AMR. As such, all of the components that comprise these alternate flow paths should be within the scope of license renewal and subject to an AMR per the requirements of 10 CFR 54.4(a)(2).

Reply

RG&E has agreed to add the spent fuel pool makeup path from the RWST to the spent fuel pool into the scope of license renewal, and subject to aging management review. An additional sentence should be added to 2.3.3.3 of the application, noting that a makeup system from the RWST to the spent fuel pool is part of the system.

All of the components being added to the scope of license renewal in the SFP Cooling System are already included within the component groups listed in Table 2.3.3-3 of the application. Specifically:

Component Group	Passive Function	Aging Management Reference
Pipe	Pressure Boundary	Table 3.4-2 line number (236) Table 3.4-2 line number (238) Table 3.4-2 line number (239)
Pump Casing	Pressure Boundary	Table 3.4-2 line number (270) Table 3.4-2 line number (274) Table 3.4-2 line number (275)
Valve Body	Pressure Boundary	Table 3.4-2 line number (394) Table 3.4-2 line number (398) Table 3.4-2 line number (399) Table 3.4-2 line number (447) Table 3.4-2 line number (452) Table 3.4-2 line number (453)

OI 2.3.3.6-1

The applicant did not provide an adequate basis in its response to RAI 2.3.3.6-1 dated May 13, 2003, for concluding that the fire service water booster pump, piping, and valves back to the service water system were excluded from the scope of license renewal.

The staff evaluated the applicant's position concerning the jockey pump and storage tank and studied the relevant documents, including the Ginna UFSAR Section 9.5.1 and the associated SER, as well as branch technical position (BTP) 9.5-1. The staff concluded, based upon this review, that National Fire Protection Association (NFPA) 20, "Standard for the Installation of Centrifugal Fire Pumps," is endorsed by Section 6.b.6 of BTP 9.5-1, which was cited by the Ginna UFSAR as the licensing basis for the plant. The requirement for jockey pumps/pressure maintenance device is stated in Section 31(e) of the 1972 edition of NFPA 20. The 1996 edition further clarifies this requirement in Section 2-19.5 which states, "The primary or standby fire

pump shall not be used as a pressure maintenance pump." The jockey pump and storage tank, and their associated piping and valves, perform a pressure maintenance function which protects the large fire pumps from damage during low-flow, high-pressure operation and is an essential part of the fire water system. The staff therefore disagrees with the applicant's response to RAI 2.3.3.6-1 concerning the fire service water booster pump, piping, and valves back to the service water system.

Reply

RG&E agrees to include the passive pressure boundary components of the fire service water booster pump, valves, and associated piping back to the service water system into the scope of license renewal, and subject to aging management review.

All of the components being added to the scope of license renewal in the fire water system are already included within the component groups listed in Table 2.3.3-6. (Note that Table 3.4-2, line number (265) and line number (434) are additions.) Specifically:

Component Group	Passive Function	Aging Management Reference
Pipe	Pressure Boundary	Table 3.4-1, line number (20) Table 3.4-2, line number (212) Table 3.4-2, line number (214) Table 3.4-2, line number (215)
Pump Casing	Pressure Boundary	Table 3.4-1, line number (20) Table 3.4-2, line number (262) Table 3.4-2, line number (265)
Valve Body	Pressure Boundary	Table 3.4-1, line number (20) Table 3.4-2, line number (429) Table 3.4-2, line number (434)

OI 2.5-1

The staff questioned the elimination of cables M0089 and M0108 from the license renewal scope. These circuits are part of the offsite power path that brings offsite power into the safety buses. The staff therefore asked the applicant to clarify how the Ginna plant can be brought to a shutdown condition from the offsite power supply if these circuits to the safety-related shutdown buses are not included within the scope of license renewal.

In a July 11, 2003, response to staff clarification questions the applicant stated that circuits M0089 and M0108 are not relied upon to cope with, or recover from a station blackout (SBO).

The entry conditions for plant procedure ECA-0.0, "Loss of All AC Power," is the loss of bus 14 and bus 16. This procedure is not entered when bus 17 and bus 18 are lost. Upon restoration of bus 14 and/or bus 16, recovery actions are taken. These recovery actions do not rely upon bus 17 or bus 18, although they may be used if available. This procedure directs activities required to achieve shutdown conditions.

The response to the staff's question does not indicate how long Ginna can remain in a safe condition following recovery of only buses 14 and 16. The Ginna UFSAR (Section 8.3.1.1.6) indicates that buses 17 and 18, which are powered from the cables in question (M0089 and M0108), supply power to the Ginna service water pumps. The concern is that recovery of offsite power to only buses 14 and 16 following an SBO will only allow the plant to continue to operate in the hot standby or hot shutdown condition. While hot standby or hot shutdown is acceptable for plant operation during the SBO coping period, if Ginna cannot be brought to cold shutdown, recovery of buses 14 and 16 may result in only a few additional hours beyond Ginna's required 4-hour coping capability. Unavailability of condensate feedwater or other limitations could limit operation in these modes. The staff notes that recovery of the Ginna emergency diesel generators (EDGs) following an SBO would allow energization of the full complement of safety buses, including buses 17 and 18. Hot standby or hot shutdown has been accepted by the staff at some plants for non-SBO scenarios such as fire protection; however, it is not clear that the same limitations as those following an SBO event exist for the other scenarios. The applicant should identify the length of time Ginna can remain in a safe condition following recovery of only safety buses 14 and 16, and provide the justification for the acceptability of that time. The justification could refer to the staff's acceptance of comparable times for other scenarios at Ginna, evidence of the ability to repair a Ginna EDG in that time period, or comparability of that time to other staff-accepted time periods (e.g., required fuel oil supplies for the Ginna EDGs).

Reply - Note that this is a revision to our September 16, 2003 response.

RG&E has determined that cables M0089 and M0108 will be placed within the scope of license renewal, and subject to aging management review. A description of the new aging management program, which is consistent with NUREG-1801, XI.E3, to address these cables is provided in Attachment 2 - "Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements".

OI 4.2.2-1

In the June 10, 2003 letter, the applicant changed its method of determining the reference temperature for pressurized thermal shock (RT_{PTS}) value for the limiting weld, SA-847, from one that was based on the chemistry factor from Table 1 in RG 1.99, Revision 2 and 10 CFR 50.61 to one that was based on the use of the Ginna surveillance data. Two methods of determining the chemistry factor and RT_{PTS} value are identified in 10 CFR 50.61 - one method based on the amount of copper and nickel in the weld and one based on the use of surveillance data. As specified in 10 CFR 50.61(c)(2)(ii)(A) the surveillance data deemed credible according to the

criteria of paragraph (c)(2)(i) of 10 CFR 50.61 must be used to determine the material-specific chemistry factor. The applicant chose to utilize surveillance data in determining the chemistry factor but has not demonstrated that the data satisfies the credibility criteria of paragraph (c)(2)(i) of 10 CFR 50.61. The chemistry factor identified in the June 10, 2003, letter is 161.9 °F. The chemistry factor identified for this weld in the Reactor Vessel Integrity Database (RVID) is 158.7 °F, which is based on the surveillance data. Although the difference in chemistry factor calculated by the applicant and that in the RVID is small, the staff would like to review the surveillance data and methodology utilized by the applicant to determine the chemistry factor and to confirm that the results satisfy 10 CFR 50.61. The applicant is to provide the surveillance data, the detailed calculations for determining the chemistry factor from the surveillance data, and the analysis that demonstrates that the surveillance data meets the credibility criteria in 10 CFR 50.61. In addition, this analysis differs from that identified in UFSAR Section A3.1.2. The applicant is also requested to provide an update to this UFSAR Section.

Reply

The surveillance data, and calculation of chemistry factors from surveillance data, are provided in excerpts from WCAP-15885, Rev. 0, "R. E. Ginna Heatup and Cooldown Curves for Normal Operation", Chapter 8, provided as Attachment 3. The analysis demonstrating that the data meets the credibility of 10 CFR 50.61 is provided in Design Analysis DA-ME-2003-042, Revision 0, provided as Attachment 4. The UFSAR Section A3.1.2 is being updated by substituting the following for the current second paragraph: "The results of the revised PTS analysis for the limiting material have been reviewed for compliance with 10 CFR 50.61. The methodology used in PTS analysis is based on the projected neutron fluence at the end of the period of extended operation. For the intermediate and lower shell forgings, the analysis was based on Regulatory Guide 1.99, Revision 2, Position 1.1, which does not rely on plant-specific surveillance data to calculate ΔRT_{PTS}. For the circumferential weld, the analysis was based on Regulatory Guide 1.99, Revision 2, Position 2.1, which does rely on plant-specific surveillance data to calculate ΔRT_{PTS}. The analysis associated with PTS has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii) and found to be acceptable.

OI B2.1.28-1

In response to RAI B2.1.28-1, the applicant indicated that Ginna has two surveillance capsules left in the core. The current schedule is to withdraw one of the capsules during the 2003 refueling outage. At that time, the capsule will have received a fast neutron fluence of 5.25×10^{19}, more than the projected dose at 60 years of 4.85×10^{19}. Because Ginna has performed, and submitted to the NRC, a reactor vessel equivalent margins analysis, the applicant indicated that it does not plan on testing that capsule. In addition, the current plan is to leave one capsule in the reactor vessel until about 2009, at which point it will have received a fast neutron fluence equivalent to 80 years of operation. However, Item 6 in GALL XI.M31 indicates that the applicant is to withdraw one capsule at an outage in which the capsule receives a neutron fluence

equivalent to the 60-year fluence so that the capsule may be tested in accordance with the requirements of the American Society for Testing and Materials (ASTM) E-185. Therefore, the staff believes the capsule withdrawn during the 2003 refueling outage should be tested.

Testing of this capsule is important because the RT_{PTS} value in the pressurized thermal shock evaluation was determined using Ginna surveillance data. The highest capsule neutron fluence is 3.746×10^{19} n/cm^2, which is below the neutron fluence projected for the reactor vessel at the end of the period of extended operation. Testing this capsule, which has a projected neutron fluence of 5.25×10^{19} n/cm^2, will ensure that the reactor vessel will remain below the pressurized thermal shock screening criteria at the end of the period of extended operation.

Item 7 in GALL XI.M31 indicates that applicants without in-vessel capsules during the period of extended operation should use alternative dosimetry to monitor neutron fluence during the period of extended operation. Because the last capsule at Ginna is to be removed in 2009, and capsules will not be available to determine the neutron fluence during the period of extended operation, alternative dosimetry should be utilized during the period of extended operation to monitor neutron fluence.

In response to RAI clarification (C-RAI) 4.2-1, the applicant indicates, in a letter dated July 30, 2003, that the capsule withdrawn in 2003 will not be tested in accordance with Table 10 Footnote E in ASTM E-185. This footnote indicates that this capsule may be held without testing following withdrawal. ASTM E-185 provides guidance for withdrawal and testing for 40 years of operation. Based on the above discussion, the staff believes this capsule should be tested. In this clarification, the applicant also indicates that Item 7 in GALL XI.M31 is not applicable to Ginna because the applicant will be using the guidance in Item 6. Item 6 and 7 are separate guidance and they should not be substituted for each other.

Reply - Note that this is a revision to our September 16, 2003 response.

RG&E has determined that the next surveillance capsule should be withdrawn in the spring 2005 refueling outage, at which time it will have accumulated fluence slightly greater than that anticipated for 60 years of operation. Testing of this capsule will be performed prior to the time that the fluence level of the latest tested capsule, Capsule S, is exceeded (approximately 2012).

The last Ginna surveillance capsule is to be removed shortly after it accumulates a fluence equivalent to 80 years of operation. The capsule specimens are to be held without testing following withdrawal. However, it is planned that we would reinsert the dosimetry monitors, such that neutron flux could continue to be monitored during the period of extended operation.

OI B2.1.36-1

Question [Program Scope]

The applicant's program inspects locations in the thimble tube associated with geometric discontinuities or area changes along the reactor coolant flow path, such as areas near the lower core plate, the core support forging, the lower tie plate, and the vessel penetrations because these are locations that are susceptible to wear resulting from flow induced vibration. The applicant states that all thirty-six thimble tubes are within the scope of this inspection program. The staff found the scope of the program to be adequate because all thirty six thimble tubes are within scope and the inspection is performed at locations most susceptible to wear resulting from flow induced vibration. The applicant has not identified the locations on the thimble tubes and guide tubes to be inspected for SCC.

Reply - Note that these are revisions to our September 16, 2003 responses.

The entire length of each thimble tube is inspected for SCC by eddy current examination. As discussed in the response to RAI 3.2.2-1, the thimble tube inspections performed under the Thimble Tube Inspection Program are credited as the activities for managing cracking due to SCC of the guide tubes. The ID surface of the guide tubes is exposed to the same environment as the OD surface of the thimble tubes. Therefore, the condition of the thimble tubes (as determined by the results of the full-length eddy current inspections) is considered to be conservatively representative of the condition of the full length of the guide tubes, which are operated at a lower temperature.

Question [Parameters Monitored/Inspected]

The eddy current examinations determine the wall thickness of the thimble tubes, allowing an assessment of the wear, and wear rate, of each tube in each location. Eddy current examination will also be utilized to detect SCC. This is acceptable because eddy current examination has been successfully utilized to determine wall thickness and wear rate. The applicant has not identified whether the eddy current examination has been qualified to detect and size SCC.

Reply

Eddy current examination will be performed using eddy current probes that are capable of detecting axially and circumferentially-oriented flaws. The eddy current examination technique has been qualified for detection of SCC, in that Ginna operating experience indicates that OD-initiated SCC indications 20% through-wall and greater are consistently detectable. Since sizing of SCC indications from eddy current data is somewhat unreliable, remediation criteria are based on detection. Any thimble tube containing an SCC indication detected by eddy current examination is removed from service (isolated) and/or replaced. The guide tube in which the "cracked" thimble tube resides would then be considered potentially cracked, and the guide tube would be inspected using an appropriate volumetric technique, such as eddy current testing (ECT) or ultrasonic testing (UT).

Question [Detection of Aging Effects]

Thimble tube inspections are conducted using a methodology specified in a Ginna Station plant-specific procedure. This procedure requires the use of a Zetec MIZ-18 Multifrequency Eddy Current Testing System. These inspections provide indication of tube wear, and tube wear rate. This is acceptable because eddy current examination has been successfully utilized to determine wall thickness and wear rate. The applicant has not identified whether the eddy current examination has been qualified to detect and size SCC.

Reply

See above response to "Parameters Monitored/Inspected".

Question [Monitoring and Trending]

In the applicants response to F-RAI B2.1.36-1, the applicant committed to perform the thimble tube inspection at every refueling outage during the period of extended operation unless inspections on a reduced frequency can be justified by engineering evaluation. However, the applicant has not identified the frequency and the basis for the frequency of inspection for the thimble tubes and guide tubes to detect SCC.

Reply

Eddy current inspections of all 36 thimble tubes for detection of SCC indications will be performed each refueling outage during the period of extended operation, unless inspections on a reduced frequency can be justified by engineering evaluation. As discussed in the response to RAI 3.2.2-1, the thimble tube inspections performed under the Thimble Tube Inspection Program are credited as the activities for managing cracking due to SCC of the guide tubes. The ID surface of the guide tubes is exposed to the same environment as the OD surface of the thimble tubes. Therefore, the condition of the thimble tubes (as determined by the results of the full-length eddy current inspections) is considered to be conservatively representative of the condition of the full length of the guide tubes, which are operated at a lower temperature.

Question [Acceptance Criteria]

The acceptance criteria are provided in Monitoring and Trending. The acceptance criteria are acceptable because the criteria allows tubes to be replaced prior to the wear reducing the wall thickness to a size that could result in failure of the tube. However, the applicant has not identified the acceptance criteria for inspection of the thimble tube and guide tube to detect SCC.

Reply

Any SCC indication in any thimble tube detected by eddy current examination requires remediation of the thimble tube, i.e., the thimble tube is isolated and/or replaced. The guide tube in which the "cracked" thimble tube resides would then be considered potentially cracked, and

the guide tube would be inspected using an appropriate volumetric technique, such as eddy current testing (ECT) or ultrasonic testing (UT).

The stainless steel fillet weld joining the BMI guide tube to the end of each BMI penetration has been inspected every outage by a remote visual (VT-2) technique for evidence of leakage. In addition, as committed to in the response to NRC Bulletin 2003-02, RG&E will perform a bare-metal examination of the RPV lower head penetrations each refueling outage until changes to the ASME Code or industry recommendations justify a change in the examination frequency. This inspection will include the stainless steel fillet weld joining each BMI guide tube to each penetration as well as the alloy 82/182 weld between the stainless steel safe-end and the lower head penetration nozzle, and will be VT-1 quality as defined in IWA 2210 of the ASME Section XI Code.

CI 2.3.3.2-1

By letter dated June 10, 2003, in response to RAI 2.3.3.2-2, the applicant stated that the piping, valve bodies, bonnets, and pump casings that can be used to fill the component cooling surge tank from the reactor water makeup tank, shown on drawing 33013-1245, are not within the scope of license renewal. The applicant cited UFSAR Section 9.2.2.4.1.3 that describes the evaluation performed in Systematic Evaluation Program (SEP) Topic IX-3, "Station Service and Cooling Water Systems," in the final SER dated November 4, 1981. The cited evaluation does not include providing makeup water to the component cooling water system until after a postulated leak is identified and isolated and repairs are made to restore the flow path to essential equipment. The applicant also references UFSAR Section 9.2.2.2 that identifies the function of the CCW surge tank as ensuring "a continuous CCW supply until a leaking cooling line can be isolated." The applicant further identified that through proper aging management of the in-scope component cooling water system components, system leakage will be minimized and the CCW surge tank will act as the makeup source for "normal" leakage. It is the applicants position that a failure of any makeup capability other than that provided by the surge tank will not affect a safety function; therefore, the makeup capability from the reactor makeup water system is out of the scope of the Rule.

The staff cannot reconcile the applicant's response with the fact that the Ginna CLB relies upon makeup to the component cooling water system in the event of leakage during post-accident operation. The components of the makeup water supply to the component cooling water system may be required to replace system leakage necessary to maintain operation of the CCW, and as such, are within the scope of license renewal and subject to an AMR per the requirements of 10 CFR 54.4(a)(2). In a letter dated September 16, 2003, the applicant stated that the components from the reactor makeup water tank will be added to the scope of license renewal and subject to an AMR.

Reply

The piping, valve bodies, and pump casings between the reactor makeup water tank TCH15 and valve MOV 823 in the CCW system have been added to the scope of license renewal and are subject to aging management review. With the exception of the reactor makeup water pump casings, all of these components are already included within the component groups in Table 2.3.3-1 of the LRA. Specifically:

Component Group	Passive Function	Aging Management Reference
Fasteners (Bolting)	Joint Integrity	Table 3.4-1 line number (13) Table 3.4-1 line number (23) Table 3.4-2 line number (80) Table 3.4-2 line number (81)
CS Components	Pressure Boundary	Table 3.4-1 line number (13)
Pipe	Pressure Boundary	Table 3.4-1 line number (5) Table 3.4-2 line number (205) Table 3.4-2 line number (236) Table 3.4-2 line number (244)
Pump Casing	Pressure Boundary	Table 3.4-2 line number (260)
Valve Body	Pressure Boundary	Table 3.4-1 line number (5) Table 3.4-2 line number (388) Table 3.4-2 line number (390)

For the reactor makeup water pump casings, which are cast iron containing treated water, a new line item (265a) is to be added to Table 2.3.3-1 of the LRA, as follows:

Pump Casing	Pressure Boundary	Table 3.4-2 line number (265a)

Because of the added line number (265a), Table 3.4-2 changes are as follows:

Component Type	Material	Environment	AERMs	Program/Activity	Discussion
(265a) Pump Casing	Cast Iron	Treated Water-Other	Loss of Material	Periodic Surveillance and Preventive Maintenance Program	Material and environment grouping are not included in NUREG-1801. The aging management program(s) referenced are appropriate for the aging effects identified and provides assurance that the aging effects are effectively managed through the period of extended operation.

CI 2.3.3.5-1

By letter dated March 21, 2003, the staff requested that the applicant justify why a portion of the service water system piping that is not subject to an AMR connects two parallel portions of the service water system piping that are subject to an AMR at valves 4733, 4651B, and 4562B. These valves are shown as normally open on license renewal boundary drawing 33013-1250, 3-LR, at locations I2, I7, and J7 (RAI 2.3.3.5-2). Two issues were raised in the subject RAI regarding this piping.

First, this piping run has two parallel trains containing air conditioning water chiller units SCI03A and SCI03B which cool the chilled water system. Drawing 33013-1920 for the chilled water system indicates that the chilled water system cools the control room ventilation system and the components are all identified as augmented quality. Section 9.4.3 of the Ginna UFSAR states that the function of the control room ventilation system is, in part, to ensure the operability of control room components during normal operating, anticipated operational transient, and design-basis accident conditions. The staff infers that this statement applies to the cooling function of the system because the filtration and boundary integrity functions do not support control room equipment operability. UFSAR Section 6.4 states that the control room ventilation system cools the recirculated air as required using chilled water coils. LRA Sections 2.3.3.5, 2.3.3.10, and 2.3.3.15 do not provide an adequate basis for excluding the associated systems and components from an AMR. The applicant was requested to provide information identifying important-to-safety portions of the service water, chilled water, and

control room ventilation systems as SCs subject to an AMR, or to justify their exclusion from an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

By letter dated May 13, 2003, the applicant responded that those portions of the service water, chilled water, and control room ventilation systems that are subject to an AMR, in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1), are identified in the LRA. While UFSAR Sections 9.4.3 and 6.4 describe all the design functions of the control room area ventilation system, only some design functions meet the inclusionary criteria in 10 CFR 54.21(a)(1). While control room cooling via chilled water with the heat ultimately rejected to service water is the preferred method, it is not the only method and does not take into account cooling via radiant heat conduction into the surrounding building members or the cooling provided by the exchange of air through the filtration and pressure boundary equipment.

The components addressed in this question were reviewed under Item III.D.3.4, "Control Room Habitability," as part of NUREG-0737 (final docketed SER dated April 11, 1983). That review included the understanding that, under certain accident conditions, service water to the chiller units is automatically isolated, thus rendering this heat removal media ineffective. The NRC design bases inspection performed in 1997 (see NRC inspection report IR 50-244/97-201) also led to additional reviews to verify that the control room would not heat up to a temperature above acceptable limits. Additionally, plant operating experience supports the assessment that control room equipment remains functional and operable without the use of the chiller packages to condition the air. Thus, the basis for the exclusion from the scope of license renewal is that these components are not important to safety and do not perform any functions listed in the scoping criteria requirements of 10 CFR 54.4.

The staff evaluated the applicant's response to RAI 2.3.3.5-2. The staff did not identify information in the references cited by the applicant that provides the information needed to support exclusion of the piping from the scope of license renewal. While it is stated in the references that redundant isolation dampers are installed on the control room ventilation system to protect against accidental releases of toxic or radioactive gases, no information could be found in either reference to support the statement that service water to the chiller units is automatically isolated, thus rendering this heat removal media ineffective. In a meeting following receipt of the response, the applicant stated that license renewal boundary drawing 33013-1250, 3-LR, dampers 4562 and 4733 are shown as isolating automatically following a "T" signal. The staff does not agree with the applicant's assertion that closing the isolation dampers implies that control room cooling function is not required by the Ginna CLB, as the cooling function could continue in a recirculation mode when the dampers are closed. Therefore, the staff requested that the applicant provide additional references demonstrating that the Ginna CLB does not credit control room cooling using the service water system following an accident to assure the

continued operability of safety-related equipment needed for accident mitigation.

Reply

Ginna Station does not rely on forced control room cooling to maintain control room habitability in the event of an accident. There is no reference to such a cooling system in the UFSAR, Sections 3.11 and 6.4. The meeting referenced in this confirmatory item discussed the fact that Service Water System isolation valves 4663 and 4733 (not dampers 4562 and 4733) receive an automatic isolation "T" signal in the event of a Safety Injection signal, as would be received following a LOCA. This isolates cooling water to the chilled water system that normally maintains control room cooling, preventing the use of forced cooling.

Control room habitability has been reviewed twice by the NRC, via SEP Topic VI-8 (SER dated January 7, 1981) and via NUREG-0737, topic III.D.3.4 (SER dated October 18, 1982). No negative issues were identified with respect to control room cooling during either review. Since our design clearly isolates cooling to the control room following an accident, and our system was reviewed by the NRC and found acceptable, control room cooling is clearly not part of the Ginna Current Licensing Basis.

CI 2.3.3.10-1

By letter dated May 21, 2003, the applicant responded to RAI Generic HVAC-2, that "the specific cooling/heating coils and heat exchangers in question only have a pressure boundary intended function, that is, their heat transfer function is not credited in the current licensing basis." However, the staff noted that under the component group, "heat exchangers," in LRA Tables 2.3.3-9 and 2.3.3-10, both pressure boundary and heat transfer are listed as intended functions. This appears to be in contradiction with the above response and was discussed with the applicant. The applicant stated that the tables were in error and committed to make the necessary corrections.

Reply

RG&E agrees that, in LRA Tables 2.3.3-9 and 2.3.3-10, both pressure boundary and heat transfer are intended functions. Therefore, the tables are accurate as provided in the application.

CI 3.3.2.3.4-1

In LRA Section B2.1.16, "Fuel Oil Chemistry," the applicant describes its AMP to manage aging of the components exposed to the fuel oil environment. The LRA states that this AMP is

consistent with GALL AMP XI.M30, "Fuel Oil Chemistry," with exceptions regarding not adding biocides, stabilizers, or corrosion inhibitors to the fuel oil and not sampling for particles in accordance with the modified ASTM D2276 test procedure. In letters dated May 13 and June 10, 2003, the applicant, responding to the staff's request for additional information F-RAI B2.1.16-1, stated that in a review of plant-specific operating experience no evidence of oil degradation or MIC has ever been observed. Therefore, addition of biocides, stabilizers, or corrosion inhibitors has not been needed to date. Effectiveness of using fuel oil without additives will be verified by the results of periodic inspections of the fuel storage tanks. In its letter, the applicant also modified its position regarding measuring particles and applying the "clear and bright" method for determining water and particulate contamination in the diesel fuel oil. The applicant made a commitment to change its technical specifications by incorporating specific particulate testing requirements for diesel generator fuel oil in accordance with the ASTM D2276 standard or its successor, and eliminating the need for the "clear and bright" method of the ASTM D4176 standard.

Reply

RG&E has made a commitment, as provided in this submittal, to submit a Technical Specification change by the end of 2004, to incorporate specific particulate testing requirements for diesel generator fuel oil, and eliminate the need for the "clear and bright" method of the ASTM D4176 standard.

CI 3.6-1

In RAI 3.6-1, the applicant was asked to provide a description of its AMP used to detect aging effects associated with certain aging stressors. The applicant provided its response in a letter dated July 16, 2003. With regard to the splice box that was constructed in 1989 to join the existing aluminum conductor Westinghouse phase bus to the new copper conductor Unibus phase bus, the applicant stated that, "It is assumed that Penetrox was used to connect the aluminum to the copper transition piece because the Westinghouse bus was not plated at the field cut/prepared end." The applicant should confirm that Penetrox or another suitable antioxidant material was indeed used on the electrical joint mating surfaces. Although the splice box will have only 40 years of operation at the end of the license renewal period of extended operation, lack of a suitable antioxidant coating on the aluminum to copper mating surfaces could result in early failure of the electrical joint.

Reply

The note "USE PENETROX BETWEEN EXISTING ALUMINUM AND NEW COPPER BUS BARS" on Drawing D-1800-013, Revision 2 (provided as Attachment C to 9/16/03 letter) was incorporated to note "as-built" conditions, i.e., those conditions actually verified in the field. This "as-built" note provides certainty in the conformance between design requirements and field conditions.

Att. 2

INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Program Description

Most electrical cables in nuclear power plants are located in dry environments. However, some cables may be exposed to condensation and wetting in inaccessible locations, such as conduits, cable trenches, cable troughs, duct banks, underground vaults or direct buried installations. When an energized medium-voltage cable is exposed to wet conditions for which it is not designed, water treeing or a decrease in the dielectric strength of the conductor insulation can occur. This can potentially lead to electrical failure.

The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of inaccessible medium-voltage cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by moisture while energized will be maintained consistent with the current licensing basis through the period of extended operation. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205, SAND96-0344, and EPRI TR-109619.

In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as insulation resistance, polarization index, dissipation factor, and time domain reflectometry.

As stated in NUREG/CR-5643, "The major concern with cables is the performance of aged cable when it is exposed to accident conditions." The statement of considerations for the final license renewal rule (60 Fed. Reg. 22477) states, "The major concern is that failures of deteriorated cable systems (cables, connections, and penetrations) might be induced during accident conditions." Since they are not subject to the environmental qualification requirements of 10 CFR 50.49, the electrical cables covered by this aging management program are either not exposed to harsh accident conditions or are not required to remain functional during or following an accident to which they are exposed.

Evaluation and Technical Basis

1. *Scope of Program:* This program applies to inaccessible (e.g., in conduit or direct buried) medium-voltage cables within the scope of license renewal that are exposed to significant moisture simultaneously with significant voltage. Significant moisture is defined as periodic exposures to moisture that last more than a few days (e.g., cable in standing water). Periodic exposures to moisture that last less than a few days (i.e., normal rain and drain) are not

significant. Significant voltage exposure is defined as being subjected to system voltage for more than twenty-five percent of the time.

At Ginna station, only 4 kV cables within the scope of license renewal and installed in underground ductbanks are within the scope of this program.

2. *Preventive Actions:* This is a surveillance testing program and no actions are taken as part of this program to prevent or mitigate aging degradation.

3. *Parameters Monitored/Inspected:* In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as insulation resistance, polarization index, dissipation factor, and time domain reflectometry.

4. *Detection of Aging Effects:* In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested at least once every 10 years. This is an adequate period to preclude failures of the conductor insulation since experience has shown that aging degradation is a slow process. A 10-year inspection frequency will provide two data points during a 20-year period, which can be used to characterize the degradation rate. The first tests for license renewal are to be completed before the period of extended operation.

5. *Monitoring and Trending:* Trending actions are not included as part of this program because the ability to trend test results is dependent on the specific type of test chosen. Although not a requirement, test results that are trendable provide additional information on the rate of degradation.

6. *Acceptance Criteria:* The acceptance criteria for each test is defined by the specific type of test performed and the specific cable tested.

7. *Corrective Actions:* An engineering evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the electrical cables can be maintained consistent with the current licensing basis. Such an evaluation is to consider the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective actions required, and the likelihood of recurrence. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other inaccessible, in-scope, medium-voltage cables.

Corrective actions are implemented at Ginna Station in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", and ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants", as committed to in Chapter 17 of the Ginna Station UFSAR and described in ND-QAP "Quality Assurance Program". Provisions for timely evaluation of adverse conditions and implementation of any corrective actions required, including root cause determinations and prevention of recurrence where appropriate, are included in the corrective action program.

Corrective actions are implemented through the initiation of an Action Report in accordance with IP-CAP-1, "Abnormal Condition Tracking Initiation or Notification (Action) Report". Equipment deficiencies are corrected through the initiation of a Work Order in accordance with A-1603.2, "Work Order Initiation".

8. *Confirmation Process:*

The confirmation process is part of the corrective action program, which is implemented in accordance with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", and ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants", as committed to in Chapter 17 of the Ginna Station UFSAR. The aging management activities required by this program would also reveal any unsatisfactory condition due to ineffective corrective action.

IP-CAP-1, "Abnormal Condition Tracking Initiation or Notification (Action) Report", includes provisions for tracking, coordinating, monitoring, reviewing, verifying, validating, and approving corrective actions, to ensure that effective corrective actions are taken. Potentially adverse trends are also monitored through the Action Report process. The existence of an adverse trend due to recurring or repetitive adverse conditions will result in the initiation of an Action Report. A-1603.6, "Post-Maintenance/Modification Testing", includes provisions for verifying the completion and effectiveness of corrective actions for equipment deficiencies. A-1603.6 provides guidance for the selection and documentation of Post-Maintenance Tests (PMTs) or Operability Tests (OPTs), guidelines to ensure equipment will perform its intended function prior to return to service, and guidelines to ensure the original equipment deficiency is corrected and a new deficiency has not been created.

9. *Administrative Controls:*

Ginna Station QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR 50, Appendix B and will continue to be adequate for the period of extended operation.

ND-PRO, "Procedures, Instructions and Guidelines" and IP-PRO-3, "Procedure Control", provide guidance on procedures and other administrative control documents. IP-PRO-3 provides guidance on procedure hierarchy and classification, content and format, and preparation, revision, review and approval of Nuclear Directives and all Nuclear Operating

Group Procedures. IP-PRO-4, "Procedure Adherence Requirements" establishes procedure usage and adherence requirements. IP-RDM-3, "Ginna Records", delineates the system for review, submittal, receipt, processing, retrieval and disposition of Ginna Station records to meet, as a minimum, the Quality Assurance Program for Station Operation (QAPSO).

10. *Operating Experience:* Operating experience has shown that XLPE or high molecular weight polyethylene (HMWPE) insulation materials are most susceptible to water tree formation. Original 4 kV plant cables installed at Ginna station are not constructed of XLPE or HMWPE insulation. The formation and growth of water trees varies directly with operating voltage. Treeing is much less prevalent in 4kV cables than those operated at 13 or 33kV. Only 4 kV cables are included in the scope of this program at Ginna station. There is no plant specific operating experience to indicate that water treeing, or a decrease in dielectric strength has occurred. As additional operating experience is obtained, lessons learned can be used to adjust the program, as needed.

References

EPRI TR-103834-P1-2, *Effects of Moisture on the Life of Power Plant Cables,* Electric Power Research Institute, Palo Alto, CA, August 1994.

EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments,* Electric Power Research Institute, Palo Alto, CA, June 1999.

IEEE Std. P1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations.*

NUREG/CR-5643, *Insights Gained From Aging Research,* U. S. Nuclear Regulatory Commission, March 1992.

SAND96-0344, *Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations,* prepared by Sandia National Laboratories for the U.S. Department of Energy, September 1996.

Att. 3

Westinghouse Non-Proprietary Class 3

WCAP-15885
Revision 0

July 2002

R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation

Westinghouse

2 FRACTURE TOUGHNESS PROPERTIES

The fracture-toughness properties of the ferritic materials in the reactor coolant pressure boundary are determined in accordance with the NRC Standard Review Plan [6]. The beltline material properties of the R.E. Ginna reactor vessel are presented in Table 1.

Best estimate copper (Cu) and nickel (Ni) weight percent values used to calculate chemistry factors (CF) in accordance with Regulatory Guide 1.99, Revision 2, are provided in Table 1. Additionally, surveillance capsule data is available for four capsules (Capsules V, R, T and S) already removed from the R.E. Ginna reactor vessel. This surveillance capsule data was also used to calculate CF values per Position 2.1 of Regulatory Guide 1.99, Revision 2 in Table 4. These CF values are summarized in Table 5. It should be noted that in addition to R.E. Ginna, surveillance weld data from Turkey Point Unit 3 and 4 and Davis-Besse was used in the determination of CF for the nozzle to intermediate shell girth weld of heat # 71249. Per WCAP-15092, Revision 3[7], the weld heat # 71249 was determined to be not credible. It should be noted here that the intermediate shell forging was determined not to be credible, while the lower shell forging and the intermediate to lower shell girth weld seam were determined to be credible.

The Regulatory Guide 1.99, Revision 2 methodology used to develop the heatup and cooldown curves documented in this report is the same as that documented in WCAP-14040, Revision 2.

TABLE 1
Summary of the Best Estimate Cu and Ni Weight Percent and Initial RT_{NDT} Values for the R.E. Ginna Reactor Vessel Materials

Material Description	Cu (%)[a]	Ni(%)[a]	Initial RT_{NDT}[b]
Closure Head Flange	n/a	n/a	-75°F
Vessel Flange	n/a	n/a	-52°F
Nozzle Shell Forging 123P118[a]	0.07	0.68	30°F
Intermediate Shell Forging 125S255[a]	0.07	0.69	20°F
Lower Shell Forging 125P666[a]	0.05	0.69	40°F
Nozzle to Intermediate Shell Girth Weld (Heat # 71249)[c]	0.23	0.59	10°F
Intermediate Shell to Lower Shell Girth Weld (Heat # 61782)[d]	0.25	0.56	-4.8°F
Ginna Surveillance Weld (Heat # 61782)[e]	0.23	0.53	---

Notes:

(a) The Cu & Ni for the forgings were taken from material, WCAP-7254[8], WCAP-14684[9] or RVID2. For the inter. & lower shell forgings, RVID2 has 0.68 Ni, however, the material cert. has 0.69 Ni. Thus, the higher Ni value will be used in the calculations. The nozzle forging copper value was not reported on the material cert, thus, per RG. 1.99 one should assume 0.35 unless justification is provided. Since the nozzle forging is made from the same material as the inter. & lower shell forgings at the same time period, it is a safe assumption to say the copper value would equal the highest Cu value from the known forgings on the vessel (i.e. 0.07).

(b) The Initial RT_{NDT} values are measured values unless otherwise noted. The values were obtained from RVID2 or WCAP-14684[9].

(c) The nozzle shell to inter. shell girth weld was fabricated from weld wire heat # 71249 Linde 80, flux Lot 8445. This is the exact heat and flux as the inter. to lower shell girth weld on the Turkey Point Units 3 & 4 Reactor Vessel. It is also identical to the Turkey Point 3 & 4 Surv. weld material. The best estimate Cu & Ni was taken from WCAP-15092 R.3[7]. This differs from RVID2 for Ginna but not for Turkey Pt. (Ref. BAW 2325).

(d) The intermediate shell to lower shell girth weld was fabricated from weld wire heat number 61782 Linde 80, flux Lot 8350. This is the same heat surveillance weld material (flux lot is 8346), but differ flux lot. The best estimate Cu & Ni was taken from WCAP-14684[9] and RVID2.

(e) The Ginna surveillance weld best estimate average Cu & Ni was determined from one unirradiated sample[8 or 10], one irradiated sample from Cap. T (Specimen W26)[10], two irradiated samples from reconstituted specimens Cap. T, and 4) Eight irradiated samples from Cap. S (W22, W23, W27, W28, W32, W35, W36 and W37)[9].

The chemistry factors were calculated using Regulatory Guide 1.99 Revision 2, Positions 1.1 and 2.1. Position 1.1 uses the Tables from the Reg. Guide along with the best estimate copper and nickel weight percents. Position 2.1 uses the surveillance capsule data from all capsules withdrawn to date. The fluence values used to determine the CFs in Table 4 are the calculated fluence values at the surveillance capsule locations. Hence, the calculated fluence values were used for all cases.

In order to account for operating temperature differences, the measured ΔRT_{NDT} values from the Turkey Point Units 3 and 4 and Davis-Besse surveillance weld data (heat #71249) were adjusted so it can be applied to the R.E. Ginna nozzle to intermediate shell girth weld, which is of the same heat. No adjustment for chemistry was necessary since the overall best estimate Cu and Ni for heat 71249 is higher than the surveillance specimen average Cu and Ni (i.e. a ratio less than 1.0)[7]. The measured ΔRT_{NDT} values from the intermediate to lower shell girth weld, which is contained in the R.E. Ginna surveillance program, were adjusted for chemistry using the ratio procedure given in Position 2.1 of Regulatory Guide 1.99, Revision 2. See Table 2 below for the Tcold operating temperatures at R.E. Ginna, Turkey Point and Davis-Besse.

TABLE 2
Inlet (Tcold) Operating Temperatures

Ginna	Turkey Point	Davis Besse
549°F (Average)*	546°F (Average) [7]	555°F (Average) [7]

* Average of the Tcold values for each Capsule removed to date for R.E. Ginna documented in the E900 Database. Current Tcold is 531°F, which started in 1996. This value will be used in future evaluations with subsequent capsule withdrawals

Contained in Table 3 are the calculated fluence values for the four capsules removed from the Ginna reactor vessel to date. Also included are the Calculated fluence values for Turkey Point Units 3 and 4 and Davis Besse. The Ginna fluence values are documented in Section 3 of this report. They were determined using ENDF/B-VI cross-sections and followed the guidance in Regulatory Guide 1.190[11]. The best available fluence information for Turkey Point Unit 3 and 4 comes from WCAP-14044[12]. The calculated fluences in WCAP-14044 were determined using ENDF/B-IV cross-sections. Thus, for conservatism the calculated fluences were increased 15% to account for going to ENDF/B-VI. [Note that WCAP-15092 Rev. 3 used the higher measured fluences from WCAP-14044] The Davis Besse material was partially irradiated at Turkey Point and Davis Besse. Per Turkey Point document PTN-ENG-SESJ-99- 0118 Revision 01 (which is a reference in WCAP-15092 Rev. 3), the cumulative measured fluence was 2.57 E19 n/cm^2. Since this too was determined using ENDF/B-IV cross-sections, then it will be increased 15% as well. This should be conservative since 15% times the calculated fluences from WCAP-14044 will not increase above the measured fluences.

TABLE 3
Calculated Integrated Neutron Exposure of the Surveillance Capsules
@ R.E. Ginna, Turkey Point Units 3 & 4, Davis-Besse

Capsule	Fluence
Ginna Capsule Fluences	
V	5.87×10^{18} n/cm^2, (E > 1.0 MeV)
R	1.02×10^{19} n/cm^2, (E > 1.0 MeV)
S	1.69×10^{19} n/cm^2, (E > 1.0 MeV)
T	3.64×10^{19} n/cm^2, (E > 1.0 MeV)
Turkey Point Unit 3 Capsule Fluences[(a)]	
T	6.99×10^{18} n/cm^2, (E > 1.0 MeV)
V	1.484×10^{19} n/cm^2, (E > 1.0 MeV)
Turkey Point Unit 4 Capsule Fluences[(a)]	
T	6.73×10^{18} n/cm^2, (E > 1.0 MeV)
Davis Besse- Turkey Point Unit 3 Combined Capsule Fluences[(b)]	
V	2.956×10^{19} n/cm^2, (E > 1.0 MeV)

NOTES:

(a) Per WCAP-14044 and increased 15% to account for ENDF/B-VI Cross-sections.

(b) Per Turkey Point document PTN-ENG-SESJ-99- 0118 Revision 01[7], the fluence is combined between Turkey Point and Davis Besse equivalent to 2.57 E19. This value was also increased 15% to account for ENDF/B-VI cross-sections.

TABLE 4

Calculation of Chemistry Factors using R.E. Ginna, Turkey Point & Davis Besse Surveillance Capsule Data

Material	Capsule	Capsule f[a]	FF[b]	ΔRT_{NDT}[c]	FF*ΔRT_{NDT}	FF^2
Lower Shell Forging 125P666	V	0.587	0.851	25	21.275	0.724
	R	1.02	1.006	25	25.150	1.012
	T	1.69	1.144	30	34.320	1.309
	S	3.64	1.335	42	56.070	1.782
				SUM:	136.815	4.827
	$CF_{LSF\ 125P666} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (136.815) \div (4.827) = 28.3°F$					
Intermediate Shell Forging 125S255	V	0.587	0.851	0	0	0.724
	R	1.02	1.006	0	0	1.012
	T	1.69	1.144	0	0	1.309
	S	3.64	1.335	60	80.1	1.782
				SUM:	80.1	4.827
	$CF_{ISF\ 125S255} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (80.1) \div (4.827) = 16.6°F$					
Ginna Surveillance Weld Metal (Heat # 61782)	V	0.587	0.851	149.8 (140)	127.480	0.724
	R	1.02	1.006	176.6 (165)	177.660	1.012
	T	1.69	1.144	160.5 (150)	183.612	1.309
	S	3.64	1.335	219.4 (205)	292.899	1.782
				SUM:	781.651	4.827
	$CF_{HL\ \#61782} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (781.651) \div (4.827) = 161.9°F$					
Turkey Point Surveillance Weld Material[d] (Heat # 71249)	Davis	2.956	1.287	221 (215)	284.427	1.656
	T (TP3)	0.699	0.900	163 (166)	146.700	0.810
	V (TP3)	1.484	1.109	176 (179)	195.184	1.230
	T(TP4)	0.673	0.889	208 (211)	184.912	0.790
				SUM:	811.223	4.486
	$CF_{Ht\ \#71249} = \Sigma(FF * RT_{NDT}) \div \Sigma(FF^2) = (811.223°F) \div (4.486) = 180.8°F$					

See Next Page for Notes

Notes:

(a) f = fluence. See Table 3, (x 10^{19} n/cm^2, E > 1.0 MeV).

(b) FF = fluence factor = $f^{(0.28 - 0.1*\log f)}$.

(c) ΔRT_{NDT} values are the measured 30 ft-lb shift values taken from the following documents:
- Ginna Plate and Weld... WCAP-14684[9].
- Turkey Point & Davis Besse... WCAP-15092 R.3[7].

(d) Ginna operates with an average inlet temperature of approximately 549°F, Turkey Point 3&4 operate with an average inlet temperature of approximately 546°F, and Davis Besse operates with an average inlet temperature of approximately 555°F. The measured ΔRT_{NDT} values from the Turkey Point 3&4 surveillance program were adjusted by subtracting 3°F to each measured ΔRT_{NDT} and the Davis Besse surveillance program data was adjusted by adding 6°F to the measured ΔRT_{NDT} value before applying the ratio procedure. The surveillance weld metal ΔRT_{NDT} values have been adjusted by a ratio factor of:
Ratio Ginna = 1.07, Ratio Turkey Point = 1.0 (conservative), Ratio Davis Besse = 1.0 (conservative)
The pre-adjusted values are in parenthesis.

TABLE 5
Summary of the R.E. Ginna Reactor Vessel Beltline Material Chemistry Factors

Material	Chemistry Factor	
	Position 1.1	Position 2.1
Nozzle Shell Forging 123P118	44.0°F	---
Intermediate Shell Forging 125S255	44.0°F	16.6°F
Lower Shell Forging 125P666	31.0°F	28.3°F
Nozzle to Intermediate Shell Girth Weld (Heat # 71249)	167.6°F	180.8°F[(a)]
Intermediate Shell to Lower Shell Girth Weld (Heat # 61782)	170.4°F	161.9°F
Ginna Surveillance Weld Seams (Heat # 61782)	158.9°F	---

(a) Using Surveillance Data from Turkey Point.

Att. 4

EVALUATION OF R. E. GINNA REACTOR VESSEL MATERIAL SURVEILLANCE DATA CREDIBILITY

Ginna Station

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DA - ME - 2003 - 042

Revision (0)

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Independent Reviewer — Date

REVISION STATUS SHEET

Revision Number	Affected Sections	Description of Revision
0	All	Initial issue

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Evaluation of R. E. Ginna Reactor Vessel Materials Surveillance Data Credibility

1.0 Purpose

1.1 Background

Due to RG&E's application for license renewal of the R. E, Ginna Nuclear Station. it was necessary to ascertain the effects of irradiation on the fracture toughness of limiting belt-line materials of the reactor vessel during the period of extended operation. Fluence values were recalculated using techniques and methodology that comply with Reg. Guide 1.190 (Reference 4.1). Reference 4.2 documents the results and technical details of this reanalysis.

RG&E determined effects of neutron radiation embrittlement on the low-alloy steels that comprise the belt-line of Ginna reactor vessel, utilizing surveillance capsules that were inserted in the reactor vessel (RV) prior to initial plant startup. Six capsules were positioned in the vessel between the thermal shield and vessel wall as shown in Figure 1. The vertical center of the capsules is opposite the vertical center of the fuel core. Reference 4.5 documents the Ginna Reactor Vessel Radiation Surveillance Program. This program complies with requirements of General Design Criterion 31. "Fracture Prevention of Reactor Coolant Pressure Boundary", of Appendix A. "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50.

Of the six capsules that had been utilized in the surveillance program. four had been pulled out and tested. Results were used to generate heat-up and cool-down pressure-temperature limit curves for Ginna up to end-of-life (EOL) after 40 years of normal operation. Reference 4.3 provides the methodology and results on the use of the surveillance data from the four tested capsules in supporting Ginna's 40 years of normal operation. The fluence values used in Reference 4.3 were based on "best estimate" methodology that is different from the one used in Reference 4.2.

The recalculated fluence values using Reg. Guide 1.190 (Reference 4.1) were utilized to determine the adjusted reference temperature of the RV belt-line materials during the period of extended operation. These fluence values were validated through comparison with measured data that are available from the four surveillance capsules that have been withdrawn and tested to date. The validation is documented in Section 3.5 of Reference 4.2.

According to the requirements of Reg. Guide 1.99 (Reference 4.4). credibility of the surveillance data have to be evaluated utilizing five criteria. The recalculated fluence values are slightly different from the values that had been previously utilized to establish credibility of the surveillance data.

1.2 Objective of Evaluation

Utilizing the recalculated fluence values, this analysis will perform a credibility evaluation of the existing surveillance data sets considering the five criteria in Reg. Guide 1.99 (Reference 4.4). Establishing credibility of the surveillance data qualifies the Ginna surveillance program to support reactor vessel operation during period of extended operation.

2.0 Conclusion

This evaluation showed that the Ginna RV material surveillance data satisfied all five criteria for establishing credibility imposed by Reg. Guide 1.99, Rev. 2 (Reference 4.4). Consequently, the data from the four capsules that had been withdrawn and tested can be utilized to establish pressure-temperature limits of Ginna operation during period of extended operation.

3.0 Design Input

3.1 Calculated surveillance capsule fluence exposures and the corresponding measured ΔRT_{NDT} (adjustments in reference temperature caused by neutron irradiation) come from Reference 4.2. These are listed in Tables 1, 2, and 3 for each belt-line material.

3.2 The credibility criteria were taken from Reference 4.4.

Table 1
Calculated Surveillance Capsule Exposure and Measured ΔRT_{NDT} Values for Lower Shell Forging 125P666

Capsule	Irradiation Time (EFPY)	Fluence (E>1.0Mev, 10^{19} n/cm^2	Measured ΔRT_{NDT} °F
V	1.4	0.587	25
R	2.6	1.02	25
T	6.9	1.69	30
S	17.0	3.64	42

Table 2
Calculated Surveillance Capsule Exposure and Measured ΔRT_{NDT} Values for Intermediate Shell Forging 125S255

Capsule	Irradiation Time (EFPY)	Fluence (E>1.0Mev, 10^{19} n/cm^2)	Measured ΔRT_{NDT} $^\circ$F
V	1.4	0.587	0
R	2.6	1.02	0
T	6.9	1.69	0
S	17.0	3.64	60

Table 3
Calculated Surveillance Exposure and Measured ΔRT_{NDT} Values for Ginna Surveillance Weld Metal (Heat # 61782)

Capsule	Irradiation Time (EFPY)	Fluence (E>1.0Mev, 10^{19} n/cm^2)	Measured ΔRT_{NDT} $^\circ$F (*)
V	1.4	0.587	149.8
R	2.6	1.02	176.6
T	6.9	1.69	160.5
S	17.0	3.64	219.4

* Values listed here are adjusted by multiplying measured values by ratio of chemistry factor for vessel weld to that of surveillance weld.

4.0 Referenced Documents

4.1 Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001.

4.2 WCAP - 15885, Rev. 0, "R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation," July 2002.

4.3 WCAP - 14684, "R. E. Ginna Heatup and Cooldown Limit Curves for Normal Operation," June 1996.

4.4 Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials," US Regulatory Commission, May 1988.

4.5 WCAP - 7254, "Rochester Gas and Electric Robert E. Ginna Unit 1 Reactor Vessel Radiation Surveillance Program," May 1969.

4.6 Westinghouse Report FP-RA-1, "Analysis of Capsule V from the Rochester Gas and Electric R. E. Ginna Unit 1 Reactor Vessel Radiation Surveillance Program," April 1, 1973.

4.7 WCAP - 8421, "Analysis of Capsule R from the Rochester Gas and Electric Corporation R. E. Ginna Unit 1 Reactor Vessel Radiation Surveillance Program," November 1974.

4.8 WCAP - 10086, "Analysis of Capsule T from the Rochester Gas and Electric Corporation R. E. Ginna Unit 1 Reactor Vessel Radiation Surveillance Program," April 1982.

4.9 WCAP - 13902, "Analysis of Capsule S from the Rochester Gas and Electric Corporation R. E. Ginna Unit 1 Reactor Vessel Radiation Surveillance Program," April 1993.

4.10 L. S. Marks, T. Baumeister, "Standard Handbook for Mechanical Engineers," Seventh Edition (Page 2-33).

4.11 Ginna UFSAR, Section 5.3.3.2

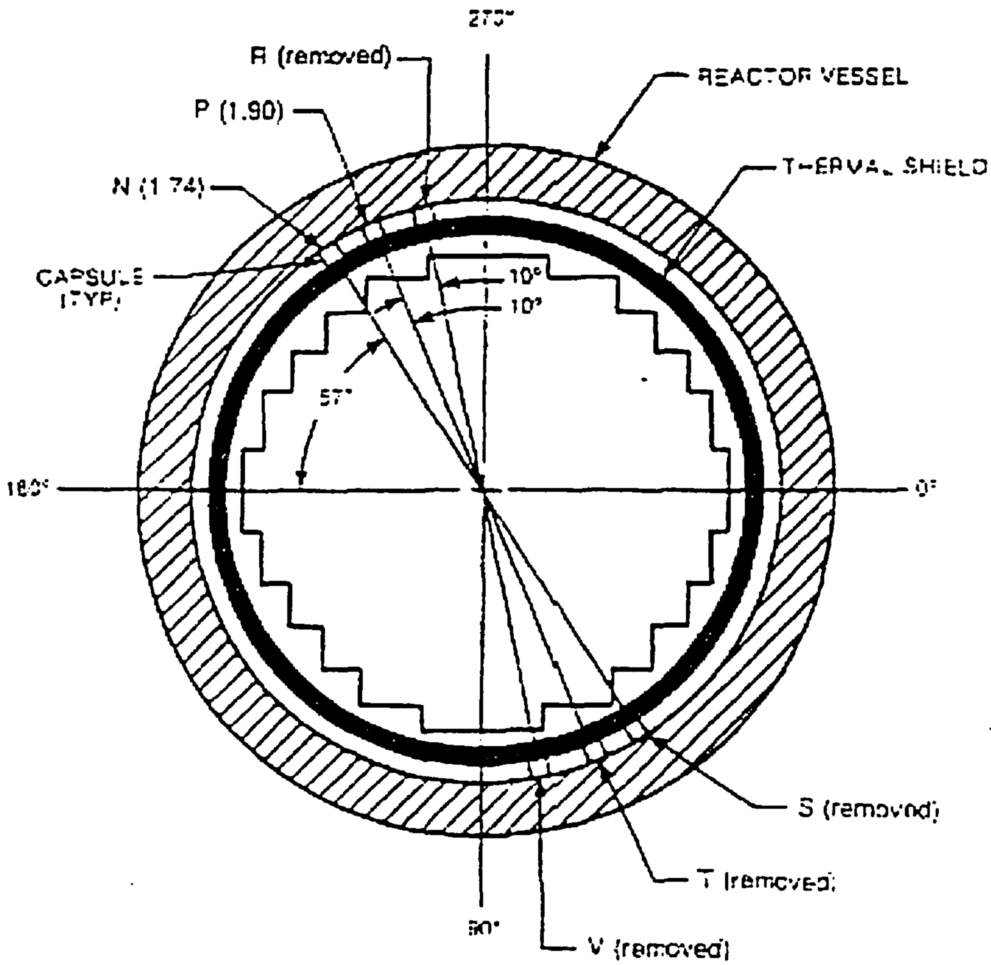
5.0 Assumptions

In this analysis, assumptions are documented and justified at sections where these are mentioned and utilized.

6.0 Computer Codes

None

Figure 1
Arrangement of Surveillance Capsules in the R. E. Ginna Reactor Vessel



7.0 Analysis

7.1 NRC Credibility Requirements

Regulatory Guide 1.99, Rev. 2 (Reference 4.4) documents general procedures acceptable to the NRC Staff for calculating effects of neutron radiation embrittlement of low-alloy steels currently used for light water-cooled reactor vessels. Position C.2 of this regulatory guide describes the method for calculating the adjusted reference temperature and Charpy upper-shelf energy of reactor beltline materials using surveillance capsule data. The methods of Position C.2 can only be applied when two or more credible surveillance data sets become available from the reactor in question, i.e., Ginna Nuclear Power Station.

7.2 Ginna Surveillance Data Sets

To date, Ginna has removed from the reactor vessel and consequently tested four (4) surveillance capsules. Test results are listed in Tables 1 to 3 for adjustment in reference temperature caused by neutron irradiation. These data sets will be shown to be credible due to a change in fluence values. As discussed in Section 1.1, fluence values were recalculated using the techniques and methodology that comply with Regulatory Guide 1.190 (Reference 4.1).

Credibility of the Ginna surveillance data sets is important since these are to be the basis for supporting Ginna operation during the period of extended operation. Changes in fracture toughness of Ginna beltline materials are ascertained from these data sets throughout the service life including the license renewal operating period. The fracture toughness of the limiting beltline materials are the basis for generating the pressure-temperature limit curves for normal heat-up and cool-down of the primary reactor coolant system.

7.3 Credibility Evaluation of Ginna Surveillance Data Sets

Per discussions in Reg. Guide 1.99 (Reference 4.4), NRC specified five (5) criteria that must be met for the surveillance data sets to be credible. Each criterion will now be evaluated to show that the Ginna surveillance data sets are credible.

7.3.1 Credibility Criterion Number 1

Criterion:

"Materials in the capsules should be those judged most likely to be controlling with regards to radiation embrittlement according to the recommendations of this guide."

Evaluation:

The beltline region of the reactor vessel is defined in Paragraph II.F of Appendix G to 10CFR Part 50 as:

"the region of the reactor vessel (shell material including welds, heat affected zones, and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions of the reactor vessel that are predicted to experience sufficient neutron radiation damage to be considered in the selection of the most limiting material with regards to radiation damage."

The Ginna reactor vessel consists of the following beltline region materials:

1. Intermediate shell forging heat number 125S255,

2. Lower shell forging heat number 125P666, and

3. Intermediate-to-lower shell circumferential weld seam SA-847 (fabricated with 1/8 inch Mn-Mo-Ni weld filler heat number 61782 and Linde 80 flux, lot number 8350).

The Ginna surveillance program as described in Reference 4.5, utilizes test specimens from both the intermediate and lower shell forgings. The weldment used in the surveillance program was fabricated with 1/8 inch Mn-Mo-Ni weld filler wire, heat number 61782 and Linde 80 flux, lot number 8436. All beltline materials are contained in the surveillance program, and any one of these materials can be the limiting or controlling material. Consequently, the Ginna reactor vessel material surveillance program satisfies credibility criterion number 1 of Reg. Guide 1.99 (Reference 4.4).

7.3.2 Credibility Criterion Number 2

Criterion:

"Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously."

Evaluation:

Plots of Charpy energy versus temperature for the unirradiated condition of the beltline materials in the Ginna reactor vessel material surveillance program are presented in WCAP 7254 (Reference 4.5).

Plots of Charpy energy versus temperature for the irradiated conditions of the beltline materials in the Ginna reactor vessel material surveillance program, are presented in the following Westinghouse reports:

- FP - RA -1 (Reference 4.6)
- WCAP - 8421 (Reference 4.7)
- WCAP - 10086 (Reference 4.8)
- WCAP - 13902 (Reference 4.9)

Each of the above report documents the analysis of the reactor vessel beltline material contained in the four Ginna material surveillance capsules. Each capsule was removed from the Ginna pressure vessel at scheduled intervals, based on a preset neutron radiation exposure inside the vessel. Capsule identification and corresponding exposure period are depicted in Tables 1, 2 and 3.

Review of the plots of Charpy energy versus temperature shown in reports mentioned above for the unirradiated and irradiated condition of each beltline material showed that the data scatters are small enough to permit a definitive determination of the 30 ft-lb temperature and upper shelf energy unambigously. Consequently, the Ginna material surveillance data sets meet credibility criterion number 2 of Reg. Guide 1.99 (Reference 4.4).

7.3.3 Credibility Criterion Number 3

Criterion:

"When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28° F for welds and 17° F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82."

Evaluation:

Ginna has currently four (4) data sets of surveillance data. The least-square-method will be utilized to determine a best-fit line, which in turn will be used to evaluate the scatter of ΔRT_{NDT} with the intention of comparing these with 28° F for welds and 17° F for base metal following the requirements of criterion 3.

<u>Method of Least Squares:</u>

Given n sets of data, (X_k , Y_k) with $k = 1....n$ and $n > 2$ required by criterion 3, the best-fit straight line for these data is given by (Reference 4.10),

$$Y = MX + B \qquad (1)$$

We assume further that the values of X_k are exact and Y_k are subject to error.

Values of M and B for n sets of data (X_k , Y_k) with $k = 1....n$, and the condition of minimized squares of the residuals with respect to this line, can be solved using the equations (Reference 4.10),

$$M\sum_{k=1}^{n} X_k^2 + B\sum_{k=1}^{n} X_k = \sum_{k=1}^{n} X_k Y_k \qquad (2)$$

$$M\sum_{k=1}^{n} X_k + Bn = \sum_{k=1}^{n} Y_k \qquad (3)$$

Since the coefficients of M and B in Equations (2) and (3) are known, their values are determined by solving the two equations in two unknowns. Equations (2) and (3) can be designated simply as,

$$M_1 M + B_1 B = C_1 \qquad (4)$$

$$M_2 M + B_2 B = C_2 \qquad (5)$$

Where:

$$M_1 = \sum_{k=1}^{n} X_k^2$$

$$B_1 = \sum_{k=1}^{n} X_k$$

$$C_1 = \sum_{k=1}^{n} X_k Y_k \qquad (6)$$

$$M_2 = B_1$$

$$B_2 = n$$

$$C_2 = \sum_{k=1}^{n} Y_k$$

Solve for M in Equation (4), and substitute into Equation (5), gives

$$M = (C_1 - B_1 B)/ M_1 \qquad (7)$$

$$M_2 [(C_1 - B_1 B)/ M_1] + B_2 B = C_2 \qquad (8)$$

Equation (8) gives the value of B as,

$$B = (M_1 C_2 - M_2 C_1)/(M_1 B_2 - M_2 B_1) \qquad (9)$$

The value of M can now be determined by substituting B from Equation (9) into Equation (7).

$$M = ((C_1 - B_1 (M_1 C_2 - M_2 C_1)/(M_1 B_2 - M_2 B_1))/ M_1) \qquad (10)$$

For the four sets of Ginna surveillance data, values of ΔRT_{NDT}, the mean value of the adjustment in reference temperature caused by irradiation, are measured corresponding to the fluence level, F (10^{19} n/cm^2, E>1 MeV). However, ΔRT_{NDT} is directly related to a quantity defined in Reg. Guide 1.99 (Reference 4.4) as the fluence factor, FF by the equation,

$$FF = F^{(0.28 - 0.10 \log F)} \qquad (11)$$

Criterion 3 implies that ΔRT_{NDT} values are subject to error because of the scatter limitation. In this analysis, we will take the value of FF, which is related to F by Equation (11) as exact. Consequently, the method of least squares discussed above will be applicable with the following corresponding quantities:

$$X = FF \qquad (12)$$

$$Y = \Delta RT_{NDT} \qquad (13)$$

And the best - fit line corresponding to Equation (1) is,

$$\Delta RT_{NDT} = M (FF) + B \qquad (14)$$

Equations (1) to (14) will now be applied to the Ginna surveillance data sets to evaluate the best - fit line and scatter for each beltline material in the capsules.

7.3.3.1 Lower Shell Forging 125P666

Using the data in Table 1, the coefficients in Equations (2) and (3) as depicted in Equations (4) and (5) are evaluated in Table 4 below.

Table 4
Coefficients for Lower Shell Forging 125P666

Capsule	$F^{(1)}$	$FF^{(2)}$ (X)	$\Delta RT_{NDT}{}^{(3)}$ (Y)	$FFx\Delta RT_{NDT}$ (XY)	FF^2 X^2
V	0.587	0.851	25	21.275	0.724
R	1.02	1.006	25	25.15	1.012
T	1.69	1.144	30	34.32	1.309
S	3.64	1.335	42	56.07	1.782
$\sum_{k=1}^{n=4}(\ )$		4.336	122	136.815	4.827
Coefficients		B_1, M_2	C_2	C_1	M_1

Notes: (1) F = Fluence (10^{19} n/cm^2, E>1.0 MeV)

(2) FF = Fluence Factor = $F^{(0.28 - 0.10 \log F)}$

(3)ΔRT_{NDT} values do not include the adjustment ratio procedure per Reg. Guide 1.99, Rev. 2, Position 2.1 since surveillance capsule materials are equivalent to actual beltline material.

For this material, the values of the coefficients are shown in Table and are summarized as follows:

$$B_1 = 4.336 \qquad B_2 = 4$$

$$C_1 = 136.815 \qquad C_2 = 122 \qquad (15)$$

$$M_1 = 4.827 \qquad M_2 = 4.336$$

Solving for B using Equation (9), we have,

$$B = (4.827x122 - 4.336x136.815) / (4.827x4 - 4.336x4.336)$$

$$= -8.55 \qquad (16)$$

Substituting value of B into Equation (10), we have,

$$M = (136.815 - 4.336x(-8.55)) / 4.827$$

$$= 36.024 \qquad (17)$$

Hence, the best-fit line for the lower shell forging 125P666 is,

$$\Delta RT_{NDT} = 36.024\ FF - 8.55 \qquad (18)$$

Equation (18) is used to calculate the scatter of ΔRT_{NDT} values about the best-fit line. Results are summarized in Table 5.

Table 5
Scatter of ΔRT_{NDT} Values About the Best-Fit Line for Lower Shell Forging 125P666

Capsule	FF	Measured ΔRT_{NDT} (30 ft-lb) °F	Best-Fit ΔRT_{NDT} (Eqn. 18) °F	Scatter of ΔRT_{NDT} °F
V	0.851	25	22.106	2.894
R	1.006	25	27.69	-2.69
T	1.144	30	32.66	-2.66
S	1.335	42	39.54	2.46

Discussions:

It is shown in Table 5 that all scatter values are less than 17° F, the acceptance value specified in criterion 3 for base metal. Furthermore, criterion 3 stipulated that for large fluence range, this value can be twice as large. Consequently, all data for the lower shell forging, 125P666 beltline material satisfy criterion 3 of Reg. Guide 1.99 (Reference 4.4).

7.3.3.2 Intermediate Shell Forging 125S255

Using data in Table 2 for this material, the coefficients of Equations (2) and (3) are calculated as shown in Table 6.

Table 6
Coefficients for Intermediate Shell Forging 125S255

Capsule	$F^{(1)}$	$FF^{(2)}$ (X)	$\Delta RT_{NDT}{}^{(3)}$ (Y)	$FFx\Delta RT_{NDT}$ (XY)	FF^2 X^2
V	0.587	0.851	0	0	0.724
R	1.02	1.006	0	0	1.012
T	1.69	1.144	0	0	1.309
S	3.64	1.335	60	80.1	1.782
$\sum_{k=1}^{n=4}(\ )$		4.336	60	80.1	4.827
Coefficients		B_1, M_2	C_2	C_1	M_1

Notes: (1) F = Fluence (10^{19} n/cm^2, E>1.0 MeV)

(2)FF = Fluence Factor = $F^{(0.28 - 0.10 \log F)}$

(3)ΔRT_{NDT} values do not include the adjustment ratio procedure per Reg. Guide 1.99, Rev. 2, Position 2.1 since surveillance capsule materials are equivalent to actual beltline material.

Values of coefficients are taken from Table 6 and are summarized below:

$$B_1 = 4.336 \qquad B_2 = 4$$
$$C_1 = 80.1 \qquad C_2 = 60 \qquad (19)$$
$$M_1 = 4.827 \qquad M_2 = 4.336$$

Solving for B, using Equation (9), we have,

$$B = (4.827x60 - 4.336x80.1) / (4.827x4 - 4.336x4.336)$$
$$= -113.77 \qquad (20)$$

Solve for M using Equation (7),

$$M = (80.1 - 4.336x(-113.77)) / 4.827 = 118.79 \qquad (21)$$

The best-fit line for the intermediate shell forging 125S255 is,

$$\Delta RT_{NDT} = 118.79\,FF - 113.77 \qquad (22)$$

Equation (22) is used to calculate the scatter of ΔRT_{NDT} values about the best-fit line for the intermediate shell forging 125S255, beltline material. Results are summarized in Table 7.

Table 7
Scatter of ΔRT_{NDT} Values About the Best-Fit Line for Intermediate Shell Forging 125S255

Capsule	FF	Measured ΔRT_{NDT} (30 ft-lb) °F	Best-Fit ΔRT_{NDT} (Eqn. 18) °F	Scatter of ΔRT_{NDT} °F
V	0.851	0	- 12.68	12.68
R	1.006	0	5.73	- 5.73
T	1.144	0	22.13	- 22.13
S	1.335	60	44.815	15.19

Discussion:

For a large fluence range, criterion 3 specifies an acceptance value for base metal as 34° F (twice the specified number of 17° F). All the scatter values in Table 7, particularly those in the large fluence range cases, are less than 34° F. Consequently, all data for the intermediate shell forging 125S255 beltline material satisfy criterion 3 of Reg. Guide 1.99, Rev. 2 (Reference 4.4).

7.3.3.3 Surveillance Weld Metal (Heat # 61782)

Using the data in Table 3, the coefficients in Equations (2) and (3) as depicted in Equations (4) and (5) are evaluated in Table 8 below.

Table 8
Coefficients for Surveillance Weld Metal (Heat # 61782)

Capsule	F(1)	FF(2) (X)	ΔRT_{NDT}(3) (Y)	$FF x \Delta RT_{NDT}$ (XY)	FF^2 X^2
V	0.587	0.851	149.8	127.48	0.724
R	1.02	1.006	176.6	177.66	1.012
T	1.69	1.144	160.5	183.612	1.309
S	3.64	1.335	219.4	292.899	1.782
$\sum_{k=1}^{n=4}(\ )$		4.336	706.3	781.651	4.827
Coefficients		B_1, M_2	C_2	C_1	M_1

Notes: (1) F = Fluence (10^{19} n/cm^2, E>1.0 MeV)

(2)FF = Fluence Factor = $F^{(0.28 - 0.10 \log F)}$

(3))ΔRT_{NDT} values include the adjustment ratio procedure per Reg. Guide 1.99 (Reference 4.4), Position 2.1. Measured values of)ΔRT_{NDT} were adjusted bu multiplying these with the ratio of Chemistry Factor for vessel weld to that for the surveillance weld specimens (Reference 4.2).

For the surveillance weld material, values of the coefficients shown in the table are summarized as follows:

$$B_1 = 4.336 \qquad B_2 = 4$$

$$C_1 = 781.651 \qquad C_2 = 706.3 \qquad (23)$$

$$M_1 = 4.827 \qquad M_2 = 4.336$$

Solving for B, using Equation (9), we have,

$$B = (4.827x706.3 - 4.336x781.651) / (4.827x4 - 4.336x4.336)$$

$$= 39.58 \qquad (24)$$

Substituting the value of B into Equation (10), we have,

$$M = (781.651 - 4.336x39.58) / 4.827$$

$$= 126.38 \qquad (25)$$

Hence, the best-fit line for the surveillance weld metal (Heat # 61782) is,

$$\Delta RT_{NDT} = 126.38\ FF + 39.58 \qquad (26)$$

Equation (26) is used to calculate the scatter of ΔRT_{NDT} values about the best-fit line, for the surveillance weld metal (Heat # 61782). Results are shown in Table 9 below.

Table 9
Scatter of ΔRT_{NDT} Values About the Best-Fit Line for Surveillance Weld Metal (Heat # 61782)

Capsule	FF	Measured ΔRT_{NDT} (30 ft-lb) °F	Best-Fit ΔRT_{NDT} (Eqn. 18) °F	Scatter of ΔRT_{NDT} °F
V	0.851	149.8	147.13	2.67
R	1.006	176.6	166.72	9.88
T	1.144	160.5	184.16	- 23.66
S	1.335	219.4	208.30	11.10

Discussion:

Table 9 shows that the scatter values of ΔRT_{NDT} about the best-fit line for the surveillance weld are all less than 28° F, the acceptance value of criterion 3 for welds. Consequently, all data for the surveillance weld material (Heat # 61782) representing the belt-line weld satisfy criterion 3 of Reg. Guide 1.99, Rev.2 (Reference 4.4) for credibility.

It is shown that all Ginna surveillance data sets that had been collected so far, satisfy the third credibility criterion of Reg. Guide 1.99, Rev. 2 (Reference 4.4).

7.3.4 Credibility Criterion Number 4

Criterion:

" The irradiation temperature of the Charpy specimens in the capsule should match vessel wall temperature at the cladding/base metal interface within +/- 25° F."

Evaluation:

As shown in Figure 5.3-3 of Reference 4.11 and also in Figure 1, the capsule specimens are located in the reactor vessel between the core barrel and the vessel wall, and are positioned such that the vertical center of each capsule is opposite the center of the core. The test capsules are in baskets attached to the thermal shield.

The location of the specimens in the capsules with respect to the reactor vessel beltline, provide assurance that the reactor vessel wall and the specimens experience equivalent operating conditions such that the temperatures will not differ by more than 25° F. The vessel wall and the specimens are exposed to the same flow field assuring that the Ginna surveillance data sets satisfy credibility condition number 4 of Reg. Guide 1.99, Rev. 2 (Reference 4.4).

7.3.5 Credibility Criterion Number 5

Criterion:

"The surveillance data for the correlation monitor material in the capsule should fall within the scatter of the data base for that material."

Evaluation:

The Ginna reactor vessel material surveillance program, also provided specimens from 6" thick ASTM correlation monitor material (A302 Gr. B) furnished by the U. S. Steel Corporation (References 4.6, 4.7, 4.8). Chemical analysis of this material reveals copper and nickel contents, which are given below.

$$\begin{aligned} Cu &= 0.20\ \% \\ Ni &= 0.18\ \% \end{aligned} \qquad (27)$$

These elements primarily influence the adjustments in reference temperatures of reactor vessel materials when exposed to neutron irradiation (Reference 4.4).

Three of the four capsules that had been removed and tested contained specimens of the correlation material. Test results for ΔRT_{NDT} at 30 ft - lb energy level for A302 Gr. B correlation monitor material is shown in Table 10.

Table 10
Calculated Surveillance Exposure and Measured ΔRT_{NDT} Values for Ginna Surveillance Correlation Monitor Material (A302 Gr. B)

Capsule	Irradiation Time (EFPY)	Fluence (E>1.0Mev, 10^{19} n/cm^2)	Measured ΔRT_{NDT} °F
V	1.4	0.587	90
R	2.6	1.02	95
T	6.9	1.69	140
S	17.0	3.64	No specimens

7.3.5.1 Calculate ΔRT_{NDT} of Correlation Monitor Material Using Reg. Guide 1.99, Rev. 2 Methods

Utilizing Reg. Guide 1.99 methodology, the adjustment in reference temperature due to neutron irradiation is given by the equation (Reference 4.4),

$$\Delta RT_{NDT} = [CF]\ F^{(0.28 - 0.10 \times \text{Log } F)} \qquad (28)$$

Where: CF = Chemistry Factor which is a function of Cu and Ni content

F = Fluence, 10^{19} n/cm^2, E>1.0Mev

For a Cu and Ni content of 0.20% and 0.18% respectively, Reg. Guide 1.99, Rev. 2 (Reference 4.4) gives a chemistry factor (CF) for base metal as,

$$CF_{A302\ Gr\ B} = 82 + 0.18x(102 - 82) / 0.20$$

$$= 100^{\circ} F \qquad (29)$$

Using Equations (28) and (29), the calculated ΔRT_{NDT} and the corresponding scatter are depicted in Table 11.

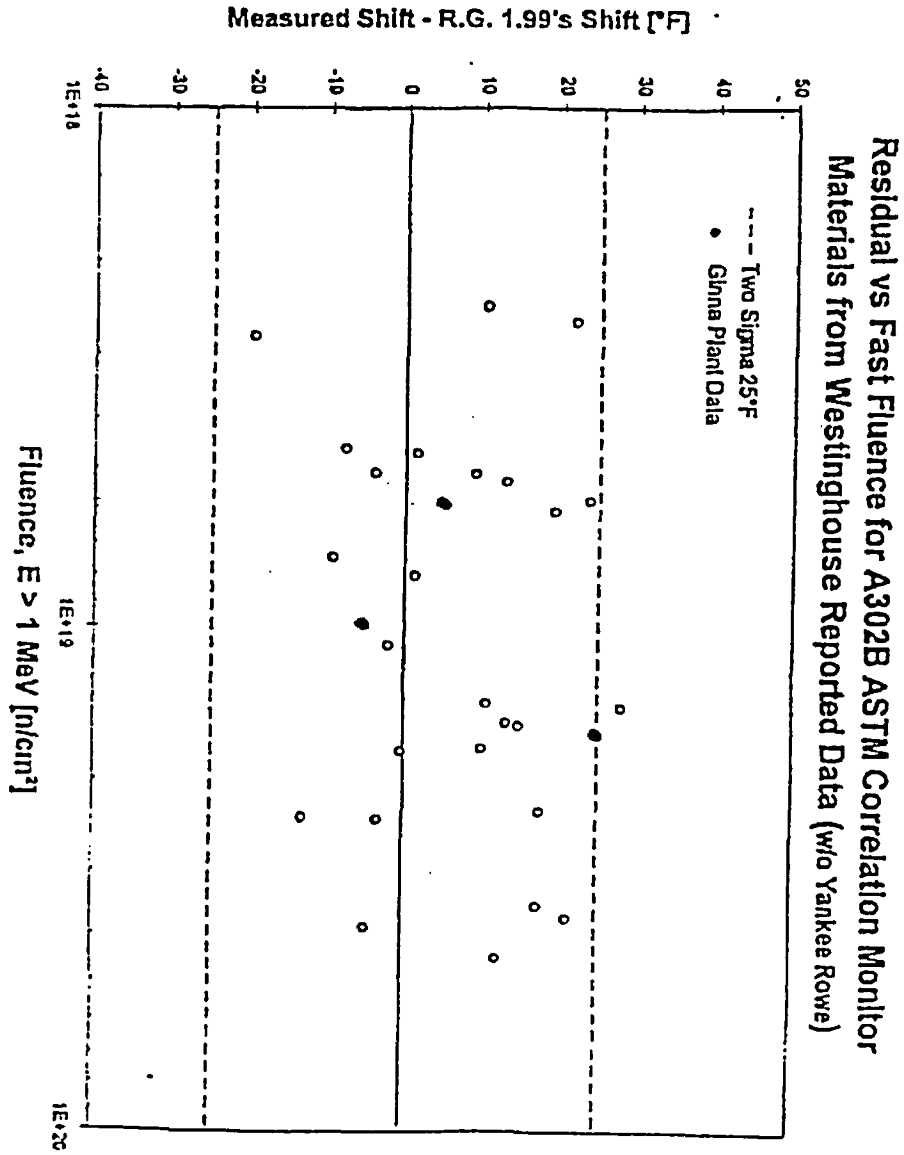
Table 11
Scatter of ΔRT_{NDT} Values About Reg. Guide 1.99, Rev2 Equation for Correlation Monitor Material (A302 Gr. B)

Capsule	Fluence (E>1.0Mev, 10^{19} n/cm^2)	Measured ΔRT_{NDT} (30 ft-lb) °F	Calculated ΔRT_{NDT} (RG1.99,rev2) °F	Scatter of ΔRT_{NDT} °F
V	0.587	90	85.1	4.9
R	1.02	95	100.6	- 5.6
T	1.69	140	114.4	25.6

The correlation monitor material, A302 Gr B that is utilized in the Ginna Reactor Vessel Material Surveillance Program was furnished by U. S. Steel Corporation through Subcommittee 11 of ASTM Committee E10 on Radioisotopes and Radiation Effects. A plot of the scatter shown in Table 11 is depicted in Figure 2, together with the data base points of A 302 Gr B. The data base points was obtained from Oak Ridge National Laboratory. The plot shows the Ginna data as solid points. The data has been shifted such that the mean value is at zero and the two-sigma band is at 25° F.

AS shown in Figure 2, all the Ginna surveillance correlation monitor material scatter data essentially fall within the two-sigma band of the data base for A302 Gr B. Consequently, the Ginna surveillance data satisfy credibility criterion number 5 of Reg. Guide 1.99, Rev. 2 (Reference 4.4).

Figure 2



8.0 Results

Evaluation results of Ginna reactor vessel material surveillance data against the five criteria for establishing credibility of Reg. Guide 1.99, Rev. 2 are summarized in Table 12.

Table 12
Results Summary for Credibility Evaluation of Ginna RV Material Surveillance Data

Reg. Guide 1.99 Criterion	Results
1. "Materials in the capsules should be those judged most likely to be controlling with regards to radiation embrittlement according to the recommendations of this guide."	The Ginna Surveillance Program utilizes test specimens taken from intermediate and lower shell forgings that were actually used as reactor vessel beltline materials. The weldment specimen representing the intermediate-to-lower shell circumferential weld seam, SA 847 was fabricated from the same 1/8" Mn-Mo-Ni weld filler wire, heat number 61782 and Linde 80 flux.
2. "Scatter in the plots of Charpy energy versus temperature for the irradiated and unirradiated conditions should be small enough to permit the determination of the 30 ft-lb temperature and upper-shelf energy unambiguously."	Plots of Charpy energy versus temperature for the unirradiated and irradiated conditions for the beltline material specimens are shown in References 4.5, 4.6, 4.7, 4.8, and 4.9. It is observed that scatter in the data are small such that the 30 ft-lb temperature and upper shelf energy are determined unambigously.
3. "When there are two or more sets of surveillance data from one reactor, the scatter of ΔRT_{NDT} values about a best-fit line drawn as described in Regulatory Position 2.1 normally should be less than 28° F for welds and 17° F for base metal. Even if the fluence range is large (two or more orders of magnitude), the scatter should not exceed twice those values. Even if the data fail this criterion for use in shift calculations, they may be credible for determining decrease in upper-shelf energy if the upper shelf can be clearly determined, following the definition given in ASTM E 185-82."	Scatters of measured ΔRT_{NDT} values for four sets of surveillance data about the best-fit line are shown below for each beltline material specimens. 1. Lower Shelf Forging 125P666 **Capsule** — **Scatter, ° F** V — 2.894 R — -2.69 T — -2.66 S — 2.46 These are less than 17° F for base metal. Hence, criterion 3 is satisfied.

	2. Intermediate Shell Forging 125S255 **Capsule** — **Scatter, ° F** V — 12.68 R — - 5.73 T — - 22.13 S — 15.19 Criterion 3 is still satisfied since for low fluence (V & R), scatter values are less than 17° F and the high fluence cases (T & S) are less than 34° F. 3. Surveillance Weld Metal (Ht. # 61782) **Capsule** — **Scatter, ° F** V — 2.67 R — 9.88 T — - 23.66 S — 11.10 Criterion 3 is satisfied since all scatter values are less than 28° F.
4. " The irradiation temperature of the Charpy specimens in the capsule should match vessel wall temperature at the cladding/base metal interface within +/- 25° F."	Locations of the surveillance capsules and specimens inside the reactor vessel are shown in Figure 1. The capsules were positioned between the thermal shield and vessel wall. The vertical center of the capsules is opposite that of the core. It is noted that the specimens and the reactor vessel wall are exposed to the same local flow field of the primary coolant inside the vessel. Consequently, the temperature of specimens in the capsule should match the vessel wall temperature at the cladding to within +/- 25° F. Hence criterion 4 is satisfied.
5. "The surveillance data for the correlation monitor material in the capsule should fall within the scatter of the data base for that material."	Scatters of measured ΔRT_{NDT} values for the correlation monitor material (A302 Gr B) about values generated using Reg. Guide 1.99, Rev. 2 (Ref. 4.4) equation are given