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June 10, 2003

U.S. Nuclear Regulatory Commission
Document Control Desk
Attn: Mr. Russell Arrighi (Mail Stop O-12D-3)
Office of Nuclear Reactor Regulation
Washington, D.C. 20555-0001

Subject: Supplemental Response to LRA Request for Additional Information (RAI)
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Arrighi:

This letter is in response to the NRC's March 21 and March 28, 2003 "Request for Additional Information for the Review of the R. E. Ginna Nuclear Power Plant, License Renewal Application". This letter supplements our May 13, May 23, and June 4 responses. Attachment 1 provides the balance of responses to the 224 RAIs. Attachment 2 provides clarifications to previous RAI responses 2.1-4, 2.3.3.13-3, 4.3.5-1, and B2.1.16-1.

The response for RAIs B2.1.15-1 and B2-1.15-2 includes data sheets provided as Attachment 3.

The clarification for RAI 4.2.1-1 and 4.2.2-1 responses includes RG&E Design Analysis DA-ME-2003-024, provided as Attachment 4. This issue was discussed in a telecon between NRC and RG&E representatives on April 23, 2003, as documented in the NRC's May 16, 2003 Summary Report.

The clarification for RAI 4.3.5-1 response includes an April 26, 1989 Structural Integrity Associates report, provided as Attachment 5.

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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 June 10, 2003

Robert C. Mecredy

Attachments

cc: w/Att. 1, 2 Mr. Russ Arrighi, Project Manager
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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. Any other statements in this submittal 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
F-RAI 3.7-3	Develop an aging management program basis document to periodically measure insulation resistance of Nuclear Instrumentation System (NIS) and High Range Radiation Monitoring (HRRM) circuits.	Prior to 9/2009
B-2.1.16-1 (clarification)	Modify Technical Specifications to incorporate specific particulate testing requirements, for diesel generator fuel oil, and eliminate use of ASTM D4176.	Prior to 9/2009

R. E. GINNA
LICENSE RENEWAL APPLICATION
REQUEST FOR ADDITIONAL INFORMATION
ATTACHMENT 1

F-RAI 2.1 -5

During the audit of the Ginna scoping and screening methodology, the audit team determined that the procedures reviewed in combination with the review of a sample of scoping and screening products provided adequate evidence that the scoping and screening process was conducted in accordance with the requirements of 10 CFR 54.4, "Scope," and 10 CFR 54.21, "Contents of Application — Technical Information." Additionally, the staff discussed the applicant's position concerning the potential long-term program implementation of the LRA methodology and guidance into the operational phase of the plant during the extended period of operation. As a result, the team concluded that the applicant needs to formally document the process it intends to implement to capture the LRA methodology and guidance upon which the applicant will rely during the period of extended operation at Ginna to satisfy the requirements of 10 CFR 54.35, "Requirements During the Term of Renewed License." The discussion should include, as appropriate, a description of the current configuration and design control processes including references to implementation guidance for those processes which are currently being reviewed for potential impact, and identification of any new process(s) or procedure(s) planned to address the integration of the LRA methodology and guidance into the operational phase of the plant.

Response

It is our intent to transition license renewal activities from the project base line phase (those activities used to develop the License Renewal Application) into the current Ginna processes. The outcome of this transition will be a process that accounts for the requirements invoked by 10 CFR 54.35 as well as 10 CFR 54.37 (b). Specifically, all plant changes, whether physical or licensing basis, will be required to account for the effects of aging on SSCs in-scope to the rule. The extent of these process changes will also require an evaluation to determine if a plant or current licensing basis change affects the scope of what is included in the license renewal aging management processes and programs.

Modification control processes will assess physical changes to the facility. These processes require completion of Change Impact Evaluation (CIE) Forms. The CIE process will be modified to ensure scoping evaluations and aging management program assignments are made, if required. Likewise the CIE process will be used to evaluate the effects on aging management program assignment when an in-scope SSC undergoes a design material change. For Licensing Basis and internal Design Analysis changes that may impact License Renewal, the Technical Input Form (TIF) will function along with the CIE to ensure the proper change evaluations are performed, including the evaluation of potential Time Limited Aging Analysis. The overall intent is to create a series of change process triggers that force an evaluation of that change for License Renewal Impact. The actual evaluation will be governed by a new procedure specific to License Renewal and similar to other process procedures in the plant Interface Procedure (IP) procedure series suite. The contents of the new procedure will be derived from the License Renewal Project procedures for Scoping, Screening and Aging Management Reviews.

With respect to programs credited with managing the effects of aging, program basis documents have been created and will be subject to the same change control processes as an internal design analysis. Maintenance and change control of programs and program basis documents will require the creation of a Nuclear Directive (ND) procedure which will establish the pedigree and quality assurance requirements.

In addition to the above, plant operating procedure changes could result in an alignment change or method of operating change which has an impact on SSCs within the scope of the rule. Accordingly, procedure change reviews will require modification account for this possibility. Finally, Current Licensing Basis (CLB) changes will need additional review screening to ensure the change does not impact the scope of License Renewal. The results of all the above reviews will be documented and maintained in a retrievable and auditable form. If any plant physical, procedural, or licensing basis change has an effect on the Updated Final Safety Analysis Report (UFSAR), the UFSAR will be updated as required by 10 CFR 50.71(e).

F-RAI 2.2 -1

LRA Table 2.2-1, "Plant Level Scoping Results," states that the systems identified below are out-of-scope, but specific components of these systems were evaluated (i.e., scoped and screened) as part of other systems for the purposes of LR:

- Plant Air
- Plant Sampling
- Circulating Water
- Fuel Handling
- Non-essential Ventilation

In addition to the systems listed above, components of the heating steam system were also evaluated as part of other systems. The heating steam system does not perform any nuclear safety function. However, localized pipe segments and equipment of the heating steam system are identified as being in the scope of LR as non safety components whose failure could prevent the satisfactory accomplishment of a safety function in accordance with 10 CFR 54.4(a)(2).

10 CFR 54.21(a)(1) states, in part, that components and their intended functions that meet the scoping criteria of 10 CFR 54.4(a) and are subject to an AMR must be identified and listed, so that their aging effects can be adequately managed consistent with the CLB. In order to confirm that SSCs with intended functions described in the UFSAR using traditional (i.e., CLB) nomenclature have been captured in the LR process, the staff needs to identify components from out-of-scope systems that were evaluated as part of the in-scope systems in the information provided in the LRA and the LR boundary drawings. Identify the components from out-of-scope systems (identified above) in the tables contained in LRA Section 2.3.

Response

It is important to note that the nomenclature used in the LRA is consistent with the UFSAR (i.e. the CLB). LRA Table 2.2-1, Plant Level Scoping Results, provides reviewers with valuable information in the "comments" column to help them understand where components are evaluated within the LRA. It was necessary to provide additional information when the UFSAR is absent that information or, more commonly, where components are required by the Standard

Review Plant to be grouped within a boundary that is not described by the plant UFSAR and CLB. System boundaries were problematic when formatting the LRA because neither NUREG 1800 nor NUREG 1801 provide meaningful descriptions of where the Staff considers one system to end and another to begin in either physical or functional terms. Consequently, we provided comments is Table 2.2-1; detailed system boundary descriptions, drawings, hypertext links between the SSCs identified Section 2 of the LRA and their corresponding aging management review in Section 3; as well as a series of systematic review tools. All of these features were designed to provide the Staff with LRA navigational waypoints of sufficient effectiveness such that the Staff could verify the LRA included and evaluated the SSCs required by 10 CFR 50.54.

The philosophy of evaluating specific components within other systems is provided in LRA Section 2.1.3. In the cases of Plant Air and Plant Sampling systems, the Containment Isolation portions of the systems were grouped in accordance with the Standard Review Plan for License Renewal Section 2.3.1 and Table 2.1-2 as well as NUREG 1801 Chapter V section C. For Non-Essential Ventilation, those portions of the system that act as fire barriers have been evaluated as a commodity, again in accordance with the standard review plan. As described in LRA Section 2.1.3, System Function Determination : "System scoping must identify all License Renewal functions associated with components contained within a system. Generally, within the License Renewal System boundary, if the system under review contains any components that meet the License Renewal scoping criteria detailed in 10 CFR 54.4(a), the entire system is considered in-scope and that system moves forward to the License Renewal screening process.

There are two specific exceptions to this dictate:

1. When the only in-scope portion of the system is comprised of components that will receive a commodity group evaluation (e.g. fire barriers, equipment supports, etc.). In this case it is appropriate to identify the system or structure as not being within the scope of License Renewal, however the basis for that determination must be clearly identified.

Example:

The Non-Essential Ventilation Systems contain components that act as fire barriers (fire dampers). Within the system evaluation boundary, no other functions performed by the system are License Renewal intended functions. Therefore, this method of evaluation of the system components that perform the fire barrier function within the Fire Barrier commodity group results in designation of the Non-Essential Ventilation Systems as not being within the scope of License Renewal.

2. When the only in-scope portion of the system is comprised of components that act as containment isolation boundaries. In that case it is appropriate to identify the system as not being within the scope of License Renewal so long as the components that perform the isolation boundary function are evaluated within the Containment Isolation Boundary System.

Example:

The Plant Sampling System contains components that act as containment isolation boundaries (valves, pipe). Within the system evaluation boundary no components, other than those that perform the isolation function, perform any additional License Renewal intended functions.

Therefore, this method of evaluation of the system components that perform the containment isolation boundary function within the Containment Isolation System results in the designation of Plant Sampling as not being within the scope of License Renewal.

Components of the specific systems addressed in this RAI are as follows:

- For Plant Air the effected components are addressed in LRA Section 2.3.2.5, Containment Isolation Components. The components are shown between the safety class 2 flags bounding the containment penetrations on drawings 33013-1882-LR; 33013-1884,1-LR; 33013-1884,2-LR; 33013-1886,2-LR; and 33013-1893-LR (note on this drawing the appropriate components are not highlighted, this is a drafting error). The affected components are pipe, valve bodies and flanges as listed in Table 2.3.2-5.

- For Plant Sampling the affected components are addressed in LRA Section 2.3.2.5, Containment Isolation Components. The components are shown between the safety class 2 flags bounding the containment penetrations on drawings 33013-1278,1-LR and 33013-1279-LR. The affected components are pipe, valve bodies, delay coil and flanges as listed in Table 2.3.2-5.

- For Fuel Handling the affected components are addressed in LRA Section 2.3.2.5, Containment Isolation Components. The components are shown between the safety class 2 flags bounding the containment penetration on drawing 33103-1248-LR and are associated with the fuel transfer slot containment penetration. The affected components are pipe, valve bodies and flanges as listed in Table 2.3.2-5.

- For Non-Essential Ventilation Systems the affected components are addressed in LRA Section 2.3.3.6, Fire Protection. As noted in the system description fire dampers are treated within the fire protection commodity group. The affected dampers are designated with an "F" adjacent to damper identification number associated with both the Essential and Non-Essential Ventilation System (LRA sections 2.3.3.10 and 2.3.3.19). These devices are not highlighted on the drawings (unless they act with a pressure boundary function to support the host systems ductwork intended function) due to their treatment as a commodity group. Specific damper identification numbers are called out in the Fire Protection Program implementing procedures. The affected components are listed under the component group "structure" in Table 2.3.3-6 with the link to Table 3.4-1 line number 19 being appropriate to fire damper frame housings.

- The Circulating Water System and the Service Water System share certain components within the scope of License Renewal. In the application, the emergency intake from the discharge canal as well as the combined Service Water/Circulating Water discharge piping is included in the Service Water system boundary. The affected components are pipe and valve bodies as listed in Table 2.3.3-5, Service Water.

<u>F-RAI 2.3.2.3 -1</u>

Screen assemblies and vortex suppressors are normally used in the containment sump which provides water for the emergency core cooling system (ECCS) recirculation phase, and one of the intended functions is to protect the ECCS pumps from debris and cavitation due to harmful vortex following a loss-of-coolant-accident (LOCA) (refer to Ginna UFSAR Section 5.4.5.4.3). Explain why the subject components were not identified as within scope in Table 2.3.2-3 of the

LRA, which listed component groups for the RHR that require an AMR.

Response

The sump screens were not included in Table 2.3.2-3 of the LRA because they are considered civil/structural components rather than ECCS system components. The screens are within the scope of the rule and are evaluated within the Containment Structure. LRA section 2.4.1 provides a description confirming their inclusion. The screen is manufactured from stainless steel and as such is evaluated within the commodity group asset CV-SS(SS)-INT as described in Table 2.4.1-1. The Residual Heat Removal System design does not employ mechanical vortex suppressors. UFSAR section 5.4.5.4.3 describes the instrumentation used to verify vortexing has not occurred during reduced RCS inventory operations.

F-RAI 2.3.3.2 -2

Section 9.2.2.4 of the Ginna UFSAR describes that the CCW system makeup capability is adequate to accommodate normal system leakage during normal and post-accident operation. This section of the UFSAR also states that the CCW lines supplying cooling to the reactor coolant pumps are not protected from dynamic effects associated with accidents and that, if a cooling line is severed, the water stored in the surge tank after a low-level alarm, together with makeup flow, provides the operator with time to close the valves external to the containment in order to isolate the leak. The UFSAR also identifies that the CCW system functions, of cooling the residual heat removal heat exchanger and the emergency core cooling system pumps, are essential. Therefore, the staff concludes that the SSCs necessary to supply makeup water from the reactor water makeup tank to the CCW system surge tank are within LR scope pursuant to 10 CFR 54.4. However, neither Section 2.3.3.2 nor Section 2.3.3.12 of the LRA identifies these SSCs as subject to an AMR. The CCW system LR flow diagram, 33013-1245-LR, indicates that only the safety-related section of piping from valves 823 and 729 (drawing location D2) to the component cooling surge tank header is within the scope of LR. Clarify whether the non safety-related piping, valve bodies, and pump casings that are necessary to provide a pressure retaining boundary, so that sufficient flow at adequate pressure is delivered from the reactor makeup water tank to the component cooling surge tank, are included within the scope of LR and subject to an AMR or justify their exclusion.

Response

The piping, valve bodies and bonnets, 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. UFSAR Section 9.2.2.4.1.3, Loss of Component Cooling Water System describes the evaluation performed in SEP Topic IX-3, Station Service and Cooling Water Systems, final SER dated 4 November 1981. The evaluation does not include providing makeup water to the Component Cooling Water system until after the postulated leak is identified and isolated, and repairs made to restore the flow path to essential equipment. As stated in the UFSAR for this evaluation, "the normal volume in the surge tank (1000 gallons) would provide operators with about 5 min at a leak rate of 210 gpm to stop a leak from the system. It is improbable that the operator could act within this time period, and it is possible that the leak may be in an unisolable portion of the system". The section then goes on to describe how safety functions are achieved if CCW can not be recovered. Additionally, UFSAR section, 9.2.2.2, System Design and Operation, identifies the function of the CCW surge tank as "ensures a continuous component cooling water (CCW) supply until a leaking cooling line can be isolated".

It is important to note that the mechanisms that initiate the CCW leaks being addressed are event driven not age related. Those portions of the CCW system that are in scope to the rule and require aging management are, as a minimum, subject to ASME Section XI class three criteria. As such, strict leakage monitoring and repair criteria must be adhered to. These requirements prohibit long term operation of the system with unisolated leaks. And while, as identified in UFSAR Section 9.2.2.4.1.4, Component Cooling Water Surge Tank, "Makeup water to the component cooling water (CCW) system is normally supplied by the reactor makeup water system via a remotely operated valve in the auxiliary building. The makeup rate is sufficient to accommodate system leakage", the makeup addressed by this statement is not relied upon for the performance of an intended function or to maintain system operability. Plant Technical Specifications for CCW provide guidance for system operability including surveillance requirements that must be adhered to should an individual component be isolated.

It is our position that through proper aging management of the in-scope CCW system components, system leakage will be minimized and the CCW surge tank will act as the make up source for "normal" leakage. Thus, because a failure of any makeup capability other then that provided by the surge tank will not affect a safety function, the makeup capability from the reactor makeup water system is out of scope.

F-RAI 2.3.3.4 -1

Vertical ball valve 1020C, from the auxiliary building sump basement piping to the auxiliary building sump, is not shown as subject to an AMR on LR boundary drawing 33013-1272, 2-LR, at location J4. However, it is relied upon to contain radiological releases in the event of an accident. Confirm if this component is subject to an AMR. If not, justify its exclusion.

Response

Vertical ball valve 1020C is subject to an aging management review. The valve should have been highlighted on the referenced drawing. Its function, however, is not to contain radiological releases but rather to prevent backflow into the residual heat removal pump pit from the auxiliary building sump.

F-RAI 2.3.3.5 -5

Drawing 33013-1250, 1-LR, at locations A1-A4 shows that the traveling screens as not being subject to an AMR. The traveling screens perform a coarse filtration function, which protects the SW pumps and other components receiving unfiltered raw water from blockage, and are typically included within the scope of LR due to that intended function. Justify the exclusion of these components from being subject to an AMR in accordance with the requirements of 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

Response

Neither the intake tunnel nor the traveling screens are credited for the operation of the Service Water System - only the Circulating Water System. The "coarse filtration" function of the screens is not credited for the operation of the Service Water pumps - the pumps themselves are equipped with suction strainers.

The clearance around the screens and the inlet structure would provide enough flow area to allow operation of the Service Water pumps, even if the traveling screens were blocked. Further, another flowpath exists which bypasses the intake tunnel completely. Opening valve 3123B allows flow to be directed from the discharge canal to the Service Water pumps . This valve and the connecting flowpath are within the scope of License renewal.

F-RAI 2.3.3.10 -4

Section 9.4.9 of the UFSAR states that the engineered safety feature's ventilation and cooling systems include those systems that service equipment required either following an accident or to ensure safe plant shutdown. Included on the provided list of equipment and/or areas serviced by these systems are the relay room and battery rooms, located in the control building. LR boundary drawing 33013-1868-LR, however, shows that the air conditioning systems servicing the relay room and the two battery rooms are not within the LR boundary.

Justify the exclusion of the air conditioning systems servicing the relay room and the battery rooms from the scope of LR and not subject to an AMR.

Response

Although the battery and relay rooms contain SSCs which perform LR intended functions, the ventilation systems for these rooms do not have an LR-intended function. These ventilation systems are not safety-related, as described in UFSAR Section 3.11.3.5. Testing and analysis has demonstrated that the post-accident temperature rise in these rooms is not rapid, and operator response measures such as opening doors and using portable air units or fans would maintain room temperatures at acceptable levels, even if the non-safety air-conditioning units provided for these rooms did not operate. Also, as stated in UFSAR section 8.1.4.5.2, an evaluation of expected room temperatures during a station blackout was performed, per Devonrue August 1990 and December 15, 1993 analyses. It was determined in this evaluation that the equipment would remain operable even with a loss of ventilation.

F-RAI 3.4 -1

a) The containment ventilation and essential ventilation systems discussed in Section 2.3 of the LRA include neoprene (elastomer) components in the systems. Normally these systems contain elastomer materials in duct seals, flexible collars between ducts and fans, rubber boots, etc. For some plant designs, elastomer components are used as vibration isolators to prevent transmission of vibration and dynamic loading to the rest of the system. In LRA Table 3.4-1, line number (2), the applicant identified the aging effects of hardening, cracks, and loss of strength due to elastomer degradation and loss of material due to wear. In the "Discussion" column of that row, the applicant credits the One-Time Inspection (B2.1.21) and the Periodic Surveillance and Preventive Maintenance Program (B2.1.23) for managing the hardening, cracking and loss of strength aging effects. The applicant also credited the System Monitoring Program (B2.1.33) for managing the aging effect of loss of material due to wear. The staff noted that the scope of the One-Time Inspection Program as described on Pages B-38 and -39 of the LRA does not include hardening, cracking and loss of strength as the aging effects of concern and does not include components that are exposed to air and gas.

Clarify how the One-Time Inspection is utilized to manage aging effects for components included in Table 3.4-1, line number (2). Also, clarify whether both the One-Time Inspection Program and the Periodic Surveillance and Preventive Maintenance Program are used for managing these aging effects. If only one of these two programs is credited for any single component, justify why One-Time Inspection alone is adequate to manage the aging effects including a discussion of the plant specific operating experience related to the components of concern to support your conclusion.

b) The staff also noted that the program description of the Periodic Surveillance and Preventive Maintenance Program on pages B-42 and -43 of the LRA includes loss of seal and not hardening and loss of strength as the aging effects of concern. Clarify whether loss of seal includes hardening and loss of strength. In addition, provide the frequency of the subject inspection described in Sections B2.1.23 and B2.2.33 for the applicable neoprene components including a discussion of the operating history to demonstrate that the applicable aging degradations will be detected prior the loss of their intended function.

Response

a) The Periodic Surveillance and Preventive Maintenance Program (PSPM) is credited for managing aging effects such as hardening, cracking and loss of strength for elastomeric materials in ventilation systems such as duct seals, flexible collars, rubber boots, etc. The scope of the PSPM program now includes inspections of these components. Vibration dampeners were evaluated under the Component Support commodity group and are included in Table 2.4.2-12 under Component Group "CSUPP-ELAST-INT".

b) The aging effect loss of seal is identified in NUREG-1801 as applicable to elastomeric components. Loss of seal may occur as a result of changes in properties of elastomers. Changes in properties may be due to hardening and cracking mechanisms which result from prolonged exposure of elastomers to elevated temperatures (greater than 95 degrees F) and ionizing radiation fields (greater than 1E6 rads). Therefore loss of seal is a result of changes in properties which include hardening and loss of strength. As discussed in (a) above, the inspections are now included in the scope of the PSPM program and are to be performed on a 6 year frequency. This frequency will be evaluated and adjusted as necessary based upon the inspection results.

F-RAI 3.4 -2

In LRA Tables 2.3.3-9 and 2.3.3-10, the AMR results for numerous components in the containment ventilation and essential ventilation systems refer to LRA Table 3.4-1, line number (5). These components include carbon/low alloy steel that are exposed to air and gas (wetted) <140 degree F. Table 3.4-1, line number (5), credits the One-Time Inspection Program, among others, for managing aging effects of loss of material due to general, pitting, and crevice corrosion and micro-biological induced corrosion (MIC) for the internal environments of ventilation systems, the diesel fuel oil systems, and the emergency diesel generator systems and credited the System Monitoring Program for managing the aging effect of loss of material for external surfaces of carbon steel components.

The staff noted that the scope of the One-Time Inspection Program as described on pages B-38 and -39 of the LRA does not include components that are exposed to air and gas. In addition, LRA Section B2.1.21, "One Time Inspection", states that the Ginna Station One-Time Inspection Program will include measures to verify the effectiveness of an existing AMP and confirm the absence of an aging effect. The applicant is requested to clarify how the One-Time Inspection is utilized to manage aging effects for the components in these two ventilation systems that are included in Table 3.4-1, line number (5). Also clarify whether both the One-Time Inspection Program and the other AMPs are used for managing these aging effects. If only one of these aging management programs is credited for any single component, justify why One-Time Inspection alone is adequate to manage the aging effects including a discussion of the plant specific operating experience related to the components of concern to support your conclusion.

Response

Table 3.4-1 line number (5), "Components in ventilation systems" includes carbon steel fan housings, damper housings, filter housings, etc., in the Containment and Essential Ventilation systems. The temperature of these housings would be expected to be the same as that of the ambient air on either side. Therefore no condensation would be expected to occur on the housing surfaces. Therefore aging effects, if any, from exposure of carbon steel to this environment would be expected to occur very slowly. A one-time inspection will be performed on these components and the results evaluated. If these inspections reveal evidence of age-related degradation, appropriate corrective actions will be taken and the specific components will be included within the scope of the Periodic Surveillance and Preventive Maintenance Program.

F-RAI 3.5 -2

In Table 3.5-1 of the LRA, line number (2), it states that piping and fitting, valve bodies and bonnets, pump casings, tanks, tubes, tubesheets, channel head and shell (except in main steam system) shall be managed for the aging effect of loss of material due to general (carbon steel only), pitting, and crevice corrosion using the Water Chemistry Program, but the Periodic Surveillance and Preventive Maintenance Program will be used to verify corrosion is not occurring in lieu of the One-Time Inspection program. NRC position is that corrosion may occur at locations of stagnation flow conditions and that a one-time inspection of select components and susceptible locations is an acceptable method to ensure that corrosion is not occurring and that the component's intended function will be maintained during the period of extended operation. The Periodic Surveillance and Preventive Maintenance Program does not contain specific details of how this inspection will be performed. For the components listed in Table 3.5-1, line number (2) of the LRA, describe how the applicant's Periodic Surveillance and Preventive Maintenance Program inspects the piping internals to ensure that corrosion is not occurring and that the component's intended function will be maintained during the period of extended operation. Also, the applicant should describe if the selection of susceptible locations for one-time inspection locations is based on severity of conditions, time of service, and lowest design margin as recommended by NUREG-1801, AMP XI-M32.

Response

Table 3.5-1, line number (2) refers to components in secondary treated water environments in Steam and Power Conversion Systems, which at Ginna Station include Main and Auxiliary

Steam, Feedwater and Condensate, Auxiliary Feedwater and Turbine-Generator and Supporting Systems. The component types linked to line number (2) include condensing chambers, pipe, valve bodies, flow elements, pump casings, tanks, controllers, governors, and trap housings. Portions of the Feedwater and Condensate and Auxiliary Feedwater Systems contain legs of piping and valves exposed to stagnant secondary treated water. Several check valves in these stagnant legs are periodically disassembled and inspected under the Periodic Surveillance and Preventive Maintenance (PSPM) program. Plant maintenance procedures which implement these inspections will be enhanced to provide explicit guidance for detection of aging effects. Any condition requiring engineering evaluation will be addressed in accordance with the Ginna Station Corrective Action program. In addition, an engineering review of piping and components in these stagnant legs will be performed to evaluate components inspected under the PSPM program for severity of operating conditions, time of service and design margin. Components with the longest time in service, lowest design margin, and most severe operating condition will be included in the PSPM program. For additional information, see also the response to RAI B2.1.23-7.

F-RAI 3.5 -4

For the steam and power conversion systems, the Periodic Surveillance and Preventive Maintenance Program is credited with managing several aging effects although it does not contain details of how these aging effects will be managed. Explain how the Periodic Surveillance and Preventive Maintenance Program will manage the aging effects for the following components: 1) LRA Table 3.5-1, line number (4) for loss of material due to general corrosion (carbon steel only), pitting and crevice corrosion, and MIC could occur in stainless steel and carbon steel shells, tubes, and tubesheets within the bearing oil coolers (for steam turbine pumps) in the AFW system, 2) LRA Table 3.5.2, line numbers (18) and (19) for loss of heat transfer and loss of material for heat exchangers in an oil and fuel environment, and 3) LRA Table 3.5-2, line numbers (23), (47), and (64) for loss of material level glass, pump casing, and valve body in an oil and fuel environment.

Also, in Table 3.5-1, line number (4), for loss of material within the bearing oil coolers, the LRA states in the discussion column, "Consistent with NUREG-1801. The Periodic Surveillance and Preventive Maintenance Program is credited with managing all applicable aging effects." Since NUREG-1801 does not contain an approved AMP for loss of material within the bearing oil coolers, explain why the AMP is considered to be consistent with NUREG-1801.

Response

1) Table 3.5-1, line number (4) refers to the stainless steel lube oil coolers for the motor-driven and turbine-driven auxiliary feedwater pumps. These coolers are shell and tube heat exchangers. The coolers are periodically cleaned and inspected under the Periodic Surveillance and Preventive Maintenance (PSPM) program. Service water flows through the tube side of these units, and lubricating oil through the shell side. The tubes of these units are inspected by eddy current testing, which is a volumetric technique and is credited for managing aging effects such as loss of material due to pitting and crevice corrosion and MIC on both the ID and OD of the tubes.

2) Table 3.5-2, line numbers (18) and (19) refer to the lubricating oil side of the outboard bearing lube oil coolers for the motor-driven and turbine-driven auxiliary feedwater pumps. It

should be noted that the lubricating oil environment to which the cast iron housing is exposed is a benign environment and would not be expected to support corrosion of the bearing housing. The lubricating oil contained in these coolers is periodically sampled and analyzed as directed by the PSPM program. The analysis includes a full spectrum of elements which has been monitored and trended over a 10 year period. Any adverse trend in the iron content could be attributed to wear particulate or corrosion products. Such a condition would be addressed under the Ginna Station Corrective Action program and would include a determination of the origin of the iron concentration.

3) Table 3.5-2, line numbers (23), (47), and (64) refer to aluminum level glass housing, cast iron pump casing, and copper alloy valve body components exposed to a lubricating oil environment. As discussed in (2) above, the PSPM program includes analysis of the lubricating oil to which these components are exposed. The analytical results provide levels of aluminum, iron and copper present in the oil. Any adverse trend in the iron content could be attributed to wear particulate or corrosion products. Any adverse trend in aluminum or copper levels would be attributed to corrosion products. These conditions would be addressed under the Ginna Station Corrective Action program and would include a determination of the origin of the element exhibiting the adverse trend.

4) The aging management program referenced in Table 3.5-1 line number (4) is "plant specific". It is to be noted the PSPM program is a plant specific program at Ginna Station and therefore the aging management program credited for managing the effects of aging for components included in line number (4) is consistent with NUREG 1801. All of the program attributes have been compared with the program elements in NUREG 1800, Appendix A and found to be consistent with the requirements.

F-RAI 3.6 -4

In line number (7), Table 3.6-1, the applicant stated: "The Structures Monitoring Program requires periodic monitoring of ground/lake water to verify chemistry remains non-aggressive. The applicant is requested to provide the results of the ground water monitoring program, in terms of chlorides, sulfates, and pH of the ground water.

Response

The most recent samples ranged between 6 and 8 ppm chloride, 20 and 40 ppm sulfate, and a pH of 7.0.

F-RAI 3.6 -7

Line number (7) of LRA Table 3.6-2 for water-control structures states that "Ginna Station does not utilize Reg. Guide 1.127, "Inspections of Water-Control Structures Associated with Nuclear Power Plants," and that the Structures Monitoring Program and Periodic Surveillance and Preventive Maintenance Program "satisfy all the appropriate criteria and provide assurance that the intended function of water control structures will be maintained through the period of extended operation." However, the description of the Structures Monitoring Program (B2.1.32) states that it will be enhanced to be consistent with RG 1.127. Resolve this apparent discrepancy and describe the enhancements that need to be made to Ginna's Structure Monitoring Program in order to make it consistent with RG 1.127. Also describe the division of the water-control structural components between the Structures Monitoring Program and the Periodic Surveillance and Preventive Maintenance Program.

Response

The Structures Monitoring Program and the Periodic Surveillance and Preventive Maintenance (PSPM) Program are inter-related, in that the PSPM program defines the periodicity of the inspections (repetitive tasks) to be performed under the Structures Monitoring program. As inspection results of the Structures Monitoring Program are analyzed, the frequency, or extent, of the inspections may be modified. These will be reflected in the PSPM program.

It is important to note that Ginna's water control structure inspection program was developed by the Army Corps of Engineers during the Systematic Evaluation Program, SEP Topic III-3.C. Regulatory Guide 1.127 was issued well after Ginna Station was licensed, and we are not committed to its use. For example, the information requested by Regulatory Position C.1 was not compiled for the Ginna water control structures. Most of the information in Regulatory Position C.2 is also not applicable to Ginna, since these structures do not exist on the site. However, the information in C.2.a,C.2.b, and C.2.e can be applied at Ginna Station. Procedure M-92.2, "Inservice Inspection of Miscellaneous Water Control Structures at Ginna" uses RG 1.127 for guidance. We will evaluate the guidance provided in Regulatory Guide 1.127 to determine if more specific detail should be included in M-92.2.

F-RAI 3.6 -14

Line number (26) in Table 3.6-1 of the LRA is for the supports for ASME piping and components and covers the aging effect cumulative fatigue damage through a TLAA. The discussion column for this table entry states that a fatigue analysis for structures and components is not incorporated into Ginna Station's CLB. NUREG-1801 recommends aging management of cumulative fatigue for these support components. Explain how the aging effect of cumulative fatigue for supports for ASME piping and components will be managed during the period of extended operation.

Response

Consistent with the Ginna CLB (Reference 1), supports for ASME piping and components were qualified and designed to the requirements of ASME III, Subsection NF (Reference 2) and AISC Manual (Reference 3). Both codes had accounted for fatigue cyclic loads by limiting the allowable stress ranges corresponding to cycles as high as greater than 2E6 cycles which bounds the number of cycles anticipated during 60 years of operation.

The Westinghouse Owners' Group Generic Technical Report (Reference 4), which has been approved by the NRC subject to limitations which were addressed in the LRA, concluded that fatigue cumulative usage factors for supports are much less than 1.0, even when effects of the extended period of operation are included. The conclusion of the evaluation is that fatigue is not an aging effect requiring management, and consequently no aging management program is needed.

Nevertheless, RG&E inspects for aging degradation of supports, including the effects of fatigue for supports of ASME piping and components, utilizing an inspection program which is documented in References 5 and 6. This program conforms to the requirements of Subsection IWF of ASME Section XI (Reference 7).

The RG&E in-service inspection program provides a Category F-A and VT-3 examination of Class 1, 2, and 3 piping supports and supports for other safety related components. It monitors and inspects for evidence of fatigue such as deformation or structural degradation of support parts. Non-conformances are administratively controlled in accordance with Reference 8. Repair or replacement actions to mitigate the consequences of fatigue (crack initiation and growth) are specified in Section 12 of the In-service Inspection Program documented in Reference 5.

References:

1. Ginna UFSAR, Section 3.9.3.3, "Pipe Supports"
2. ASME Boiler and Pressure Vessel Code, 1974 Edition, Section III, Subsection NF
3. Manual of Steel Construction, AISC, 7th Edition
4. Westinghouse Report, WCAP-14422 Rev. 2-A, "License Renewal Evaluation: Aging Management for Reactor Coolant System Supports", December 2000.
5. RG&E In-service Inspection Program, November 2, 2001
6. RG&E Nuclear Directive, ND-IIT, "In-service Inspection and Testing"
7. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWF, 1995 Edition with 1996 Addenda
8. RG&E Procedure IP-CAP-1, "Abnormal Condition Tracking Initiation or Notification (ACTION) Report.

F-RAI 3.7 -2

Statements made in Section 3.7 and Table 3.7-1 of the LRA seem to indicate that for the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program, all the accessible cable and connections (not just samples) within the identified plant buildings/areas will be visually inspected; and the inspections will include the entire building/area and not be limited to only adverse localized environments within those buildings/areas.

Section 3.7 of the LRA, under AERM, states that thermal life was not used to determine the scope of components in the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. With regard to radiolysis and radiation induced oxidation it's also stated that the results of the review were not used to determine the scope of the components in the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program. It's further indicated in Section 3.7 that that the non-EQ cable and connection program includes all in-scope, electrical cables and connections within specified plant spaces, and adequately addresses aging effects due to thermal conditions and radiation.

In Table 3.7-1 of the LRA, under the line number (2), it states that all material/environment combinations will be included under the scope of the program using an encompassing approach. In Section B2.1.11; however, under Program Description, it's stated that selected cables and connections from accessible areas (the inspection sample) are inspected and represent, with reasonable assurance, all cables and connections in the adverse localized environments. It's also indicated in Section 3.7, under Environment, that Ginna Station has identified specific plant spaces that may lead to cables exceeding 80% of ampacity due to cable tray fill deratings; and these areas are included in the non-EQ cable and connection program.

It is not clear from the above statements whether the inspections under the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will be limited to samples within adverse localized environments, or wether all cables and connections within the designated buildings/areas will be inspected. If only a sample of all cables and connections are inspected, provide the technical basis for the sample, consistent with GALL Program XI.E1 attribute number 3 on parameters monitored/inspected. Indicate whether the sample will include the PVC cables in containment identified in line number (2) of Table 3.7-1.

The Ginna UFSAR Supplement in LRA Section A2.1.9, for the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program, indicates that inspections are made in accessible areas exposed to adverse localized environments. Based on your response to the above request this supplement may require revision.

Response

The program described in B2.1.11 has been revised and is described below. This program expands the scope of the NUREG-1801 Section XI.E1 program such that plant spaces containing electrical equipment subject to an AMR will be included within the scope of the program. This scope does not limit the program to adverse localized equipment environments, but is structured to identify any such areas that may exist within the plant space. All cables identified with high loading or less than optimal cable tray fill are installed in plant spaces included in the scope of the aging management program. Since containment is a plant space within the scope of the program, the PVC cables in containment are addressed as part of the program. Ginna Station recognizes that it is not the intent of EPRI TR-109619 or NUREG-1801 Section XI.E1 that each component within an environment must be individually examined. The aging management program allows for a graded approach to examination based on operating experience and the specific environment . Therefore it is not the intent to imply that all the accessible cable and connections within the identified plant building/areas will be visually inspected. When it is clear during the implementation of the program that a plant space contains no significant stressors and is within the analyzed assumptions for limiting materials of construction, then detailed inspections are not likely to occur. However, this does not eliminate the plant space from review for future inspections. Ginna Station has determined that the aging management program meets and exceeds the intent and guidance of NUREG-1801 Section XI.E1 and is therefore adequate for managing the effects of aging for insulated cables and connections.

Electrical Cables and Connections Not Subject to 10CFR 50.49 Environmental Qualification Requirements Program.

Program Description

The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of electrical cables and connections that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by heat, radiation, or moisture 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. Conductor insulation materials used in cables and connections may degrade more rapidly than expected in these adverse localized environments. Selected cables and connections from accessible areas are inspected and represent, with reasonable assurance, all cables and connections in the inspection area. This aging management program uses a graded approach to inspection based on operating experience and observed environmental conditions. If an unacceptable condition or situation is identified for a cable or connection in the inspection area, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections. Technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205-2000, SAND96-0344, and EPRI TR-109619 are considered.

Scope of Program

This inspection program applies to accessible electrical cables and connections within the scope of license renewal that are installed or stored in the following plant buildings/areas (inspection areas):

Auxiliary Building, Standby Auxiliary Feedwater Building, Control Building, All-Volatile –Treatment Building, Cable Tunnel, Diesel Generator Building, Intermediate Building, Reactor Containment, Service Building, Screen House, Turbine Building, Technical Support Center, Transformer Yard

Plant buildings/areas not listed above that are used to store electrical cables and connections in the scope of license renewal for a specific, approved application (i.e. Appendix R equipment restoration) do not have adverse localized environments.

Preventative Actions

This is an inspection program and no actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored/Inspected

Readily accessible non-EQ insulated cables and connections installed in the areas described in the scope of this program are visually inspected for moisture and cable and connection jacket surface anomalies such as embrittlement, discoloration, cracking or surface contamination. Cable and connection jacket surface anomalies are precursor indications of conductor insulation aging degradation from heat or radiation in the presence of oxygen and may indicate the existence of an adverse localized equipment environment. An adverse localized equipment environment is a condition in a limited plant area that is significantly more severe than the specified service condition for the insulated cable or connection.

Detection of Aging Effects

Conductor insulation aging degradation from heat, radiation, or moisture in the presence of oxygen causes cable and connection jacket surface anomalies. Accessible electrical cables within the scope of license renewal and installed in plant areas described in the scope of this program are visually inspected 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 inspection for license renewal is to be completed before the end of the current license period.

Monitoring and Trending

The two 10-year inspections will provide data that can be used to assess a trend in the degradation rate of the cables.

Acceptance Criteria

The accessible cables and connections are to be free from unacceptable, visual indications of surface anomalies, which would suggest that conductor insulation or connection degradation exists. An unacceptable indication is defined as a noted condition or situation that, if left unmanaged, could lead to a loss of the intended function.

Corrective Actions

All unacceptable visual indications of cable and connection jacket surface anomalies are subject to an engineering evaluation in accordance with the plant corrective action program. Such an evaluation is to consider the age and operating environment of the component, as well as the severity of the anomaly and whether such an anomaly has previously been correlated to degradation of conductor insulation or connections. Corrective actions may include, but are not limited to, testing, shielding or otherwise changing the environment, or relocation or replacement of the affected cable or connection. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections.

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".

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.

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).

Operating Experience

Operating Experience has shown that adverse localized environments cause by heat or radiation for electrical cables and connections may exist next to or above (within three feet of) steam generators, pressurizers or hot process pipes, such as feedwater lines. These adverse localized environments have been found to cause degradation of the insulating materials on electrical cables and connections that is visually observable, such as color changes or surface cracking. These visual indications can be used as indicators of degradation.

F-RAI 3.7 -3

The discussion in line number (3) of Table 3.7-1 indicates that the treatment, at Ginna, of non-EQ electrical cables used in instrumentation circuits that are sensitive to reduction of conductor insulation resistance is not consistent with NUREG-1801. It states that external inspection of cables and connectors and their host environments identifies the possibility of thermal aging long before instrument loop adjustments can't compensate for current leakage.

Provide evidence or operational experience that supports this statement for non-EQ radiation monitoring and nuclear instrumentation cables. Such evidence could come from non-EQ radiation monitoring and nuclear instrumentation cables in the field or following accelerated aging tests. We would be looking for examples of cables that exhibited visual signs of thermal aging, even though the current leakage of the circuits was small relative to the output signal level of the circuit. If this information is not available, the MAP (XI.E2) identified in NUREG-1801 should be adopted to ensure the aging of non-EQ radiation monitoring and nuclear instrumentation cables is appropriately managed consistent with the requirements in 10 CFR 54.21(a)(3).

Response

Based on the evidence presented in NUREG/CR-5772, RG&E has concluded that the mechanical aging effects are more pronounced than the electrical aging effects and therefore Ginna Station has determined that the visual inspection for mechanical aging effects will be more effective than attempting to implement a program such as that described in NUREG-1801 Section XI.E2. The testing described in NUREG/CR-5772 includes a type of coaxial cable that may be used in instrumentation circuits that would be sensitive to reduction of conductor insulation resistance. The summary of condition monitoring measurements (Section 3.9) states in part, "Insulation resistance, polarization index, capacitance, and dissipation factor changes with aging were observed for some materials, but they were not nearly as sensitive to aging as the mechanical measurements". Ginna Station understands that cable jacketing performs only a mechanical function and does not serve an electrical function for this type of cable. The degradation to cable jackets caused by heat and radiation is observed as cracking, discoloration, and other visually identifiable anomalies.

That being said, Ginna Station periodically performs insulation resistance testing on the Nuclear Instrumentation System circuits and High Range Radiation Monitor circuits. This testing is conducted based on plant specific operating experience and is used to identify gross changes in insulation resistance that could have an adverse impact on circuit operation. While changes in insulation resistance are sometimes caused by heat or radiation, moisture is also a stressor that may cause a reduction in insulation resistance. Ginna Station intends to continue periodic testing throughout the period of extended operation. Therefore an aging management program based on the measurement of insulation resistance has been provided below. Ginna Station considers that this program more directly addresses the aging effect identified in NUREG-1801 Section XI.E2. Use of the insulation resistance testing does not preclude visual inspections of the accessible portions of these circuits as described in response to RAI 3.7-2.

Description of the program is as follows:

Exposure of electrical cables to adverse localized environments caused by heat, radiation or moisture can result in reduced insulation resistance (IR). An adverse, localized environment is defined as a condition in a limited plant area that is significantly more severe than the specified service condition for the circuit. Reduced IR causes an increase in leakage currents between conductors and from individual conductors to ground. A reduction in IR is a concern for circuits with sensitive, low-level signals such as radiation monitoring and nuclear instrumentation since it may contribute to inaccuracies in the instrument circuit.

The purpose of this aging management program is to provide reasonable assurance that the intended function of high voltage, low signal circuits exposed to an adverse localized environment caused by heat, radiation or moisture will be maintained consistent with the current licensing basis through the period of extended operation.

In this aging management program, an appropriate test, such as an insulation resistance test, will be used to identify the potential existence of a reduction in cable IR.

Scope of Program - This program applies to electrical cables used in circuits with sensitive, high voltage, low-level signals such as radiation monitoring and nuclear instrumentation that are within the scope of license renewal.

Preventive Actions - No actions are taken as part of this program to prevent or mitigate aging degradation.

Parameters Monitored or Inspected - The parameters monitored include a loss of dielectric strength caused by thermal/ thermoxidative degradation of organics, radiation-induced oxidation (radiolysis) of organics, or moisture intrusion.

Detection of Aging Effects - Cables will be tested at least once every 10 years. Testing may include insulation resistance tests, or other testing judged to be effective in determining cable insulation condition. Following issuance of a renewed operating license, the initial test will be completed before the end of the initial 40-year license term.

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.

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

Corrective Actions - 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".

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.

Administrative Controls - The documents which implement the program are subject to administrative controls, including a formal review and approval process, are 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.

Various procedures provide the required administrative controls, including a formal review and approval process, for procedures and other forms of administrative control documents.

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)..

Operating Experience - Operating experience has shown that anomalies found during cable testing can be caused by degradation of the instrumentation circuit cable and are a possible indication of potential cable degradation. Gross changes in insulation resistance may be indicative of cable degradation caused by excessive heat, radiation, or moisture.

F-RAI 3.7 -4

The discussion in line number (3) of Table 3.7-1 of the LRA indicates that surveillance, such as calibration, may not be as good a choice as visual inspection to detect aging effects in low signal level instrumentation cable. It states that the predominate cause of non-event driven degradation in cable and connector insulation is thermal aging.

Another potential cause of cable degradation is moisture. Chapter 3 of EPRI report TR-103834-P1-2, "Effects of Moisture on the Life of Power Plant Cables," identifies some water-related problems with instrumentation type circuits. The operating experience summary states that the first problem type, affecting the noise immunity of instrumentation circuits, was due to submergence degrading the jackets of instrumentation and coaxial cables. It would appear from this statement that activities such as checking for increases in signal distortion level or other signal anomalies during the calibration process, would add additional benefit to the calibration surveillance and make it a more effective tool for detecting cable aging effects. This could be of particular benefit to the highly sensitive radiation monitoring and nuclear instrumentation circuits, on the portion of the cable run that is located in conduit, subject to moisture intrusion, and not capable of being visually checked.

Provide a description of your AMP, in accordance with the requirements of 10 CFR 54.21(a)(3), used to detect cable-in-conduit aging effects that can increase signal distortion level or other signal anomalies in non-EQ radiation monitoring and nuclear instrumentation circuits; or provide justification why such a program is not needed.

Response

Plant specific operating experience indicates that moisture intrusion was identified as a significant stressor to the nuclear instrumentation circuits. Anomalies were identified during insulation resistance testing. As a result, actions were taken consistent with TR-103834 Part 2 to install weepholes and breather screens in cable pull boxes when possible. These actions increased insulation resistance in most cases, however it was expected that most of the moisture intrusion occurred at the connector due to installation practices and materials of construction. Subsequent to these actions Ginna Station initiated a project to replace NIS cables and connectors in containment due to known aging effects and plant specific operating experience. As indicated in TR-103834 Part 2, measurements of gross changes in insulation resistance is one proven method used to identify moisture intrusion. Consistent with existing preventative maintenance practices, Ginna Station intends to continue periodic insulation resistance testing throughout the period of extended operation for Nuclear Instrumentation and High Range Radiation Monitoring circuits. Therefore an aging management program based on the measurement of insulation resistance is described in response to RAI 3.7-3.

F-RAI 3.7 -9

Section 2.1.6 of the LRA discusses the general process used during the LR integrated plant assessment at Ginna Station for each of six issues the NRC staff has identified in interim staff guidance. The treatment of electrical fuse holders is one of the issues addressed. The final staff position is under development as the staff continues discussions with NEI on this topic. If this process is not finalized in time for this issue to be addressed in the staffs' Ginna LR SER, you will be asked to provide a commitment to implement the final staff guidance on this subject at Ginna, consistent with the staff's practice on previous license renewal applications. If the final staff position is finalized in time for this issue to be addressed in the staffs' Ginna LR SER, you will need to address the position.

Response

Ginna Station has reviewed plant design documents and identified a limited number of fuse holder installations that are not part of a larger assembly. For several of the installations, a failure of the fuse (or fuse holder) does not prevent a safety function identified in 10CFR54.4(a)(1) from being accomplished. All fuse holder installations are enclosed to prevent mechanical damage and exposure to moisture or contaminants. No installations were identified that are used to routinely isolate the load device and therefore fatigue of the metallic portion of the fuse holder is considered unlikely. Additionally, none of the identified installations are subject to significant vibration, chemical contamination, or corrosion. Several of these installations were confirmed by visual inspection. Stress caused by thermal expansion and contraction of the metal is limited to the amount of current carried by the circuit and the frequency of load cycling. Only one power circuit with significant current capacity was identified that contained fuse holders that meet the intended scope of the interim staff guidance. These fuse holders were installed in 1996 as supplemental penetration protection for the pressurizer backup heater group. This heater group is infrequently energized, and would not be subject to significant thermal stress.

Ginna Station reviewed entries in the corrective action program searching for deficiencies related to fuse holders and fuse clips, and determined that there have been a limited number of failures and no failures of such components that are not part of a larger assembly. The deficiencies identified are focused on only those locations such as motor control centers and switchgear where the fuses are removed for component maintenance. All such issues were readily identified during maintenance, and did not adversely impact component function.

Ginna Station reviewed NUREG-1760 and Information Notices identified in the March 10, 2003 letter from the NRC to NEI. NUREG-1760 provides little evidence to suggest that the fuse holders at Ginna Station are subject to aging effects requiring management within the period of extended operation. Issues discussed in Information Notices do not identify a stressor applicable to the fuse installations at Ginna Station. All fuse holders identified at Ginna Station as meeting the intended scope of the interim staff guidance have been installed as part of plant modifications and are not original plant equipment. None of the fuse holders identified as within the scope of the ISG will have 40 years of accumulated life at the end of the period of extended operation.

Based on a review of industry operating experience, plant specific operating experience, plant environments, and selected visual inspections, the fuses identified at Ginna Station that meet the intended scope of the interim staff guidance do not have aging effects requiring management within the period of extended operation. Ginna Station will continue to monitor industry and plant specific operating experience for aging effects that may be applicable to components subject to Aging Management Review and take steps as necessary to mitigate applicable aging effects as they arise.

F-RAI 4.3.2 -1

Section 4.3.2 of the LRA contains a discussion of the evaluation of USA Standard B31.1 components at the Ginna Station. The LRA indicates that the USA Standard B31.1 limit of 7000 equivalent full range cycles may be exceeded during the period of extended operation for the nuclear steam supply system (NSSS) sampling system and that an engineering evaluation will be performed prior to the period of extended operation. The LRA further indicates that the effects of fatigue may be managed by an inspection program if the results of the engineering evaluation are not acceptable. The UFSAR Supplement provided in Section A3.3.3 does not discuss this option. Clarify the proposed options for addressing the NSSS sampling system and provide an update of the UFSAR Supplement, if necessary. In addition, describe the existing qualification of the NSSS sampling system and provide the maximum calculated thermal stress range for affected portions of the system.

Response

The engineering evaluation of the affected portions of the NSSS sampling system has been completed. Considering thermal loads produced when the piping is heated from ambient temperature to 650 degrees F, the existing configuration of the sampling system piping from the reactor coolant system was evaluated. The maximum thermal stress range calculated was compared to the allowable value required by ANSI B31.1 (Reference 1) for 100,000 or more cycles.

According to Reference 2, the maximum thermal stress developed in the piping system during heat-up from ambient to 650 degrees F is 4660 psi. The code allowable stress range corresponding to 100,000 or more operational cycles is 9312 psi. Since the maximum thermal stress is enveloped by the code allowable value, the existing NSSS sampling system is acceptable for the period of extended operation.

As a result of this calculation, crack initiation and growth due to fatigue is not an aging effect requiring management for the NSSS sampling system at Ginna Station. Consequently, an update of the UFSAR supplement provided in Section A3.3.3 is not needed.

References:

1. ANSI B31.1 -1973, "Power Piping"
2. RG&E Analysis, DA-ME-2003-012, Rev. 0 "Evaluation of B Hot Leg Sampling Piping for Cyclic Operation During License Renewal."

F-RAI 4.3.7 -1

Section 4.3.2 of the LRA discusses RG&E's evaluation of the impact of the reactor water environment on the fatigue life of components. The discussion references the fatigue sensitive component locations for an older vintage Westinghouse plant identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." The LRA indicates that the later environmental fatigue correlations contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," were considered in the evaluation. Provide the following information for the six component locations listed in

NUREG/CR-6260:

a) For those locations with existing fatigue analyses, provide the results of the fatigue usage factor calculation, including the calculated environmental multiplier (Fen). Show how Fen was calculated.

b) For the USA Standard B31.1 locations discussed in Section 4.3.7.3 of the LRA, describe the fatigue usage factor calculation and provide the calculated fatigue usage factor. Include a detailed comparison of the Ginna Station components with the components listed in NUREG/CR-6260 and discuss the significance of the differences. This comparison should also include any differences in the thermal sleeve designs.

Response

(a) Two of the six component locations listed in NUREG/CR-6260 have explicit fatigue analyses at Ginna Station. These are the reactor vessel shell and lower head, and the reactor vessel inlet and outlet nozzles. The original fatigue analysis for the reactor vessel was performed according to the rules of ASME Section III, Subsection NB-3600. It has been demonstrated (see response to RAI 4.3.1-1) that the 60-year projections for the number of actual plant design transient cycles are bounded by the original (40-year) design transient cycle set. The environmental fatigue multiplier for these locations was calculated (Reference 2) using the appropriate F_{en} relationships from NUREG/CR-6583 (for carbon/low alloy steel associated with the RPV locations), NUREG/CR-5704 (for stainless steel associated with the piping locations) for the material for each location. These expressions are:

For Carbon Steel: $F_{en} = \exp(0.585 - 0.00124T - 0.101S^*T^*O^*\dot{\varepsilon}^*)$
For Low Alloy Steel: $F_{en} = \exp(0.929 - 0.00124T - 0.101S^*T^*O^*\dot{\varepsilon}^*)$

where:	F_{en}	=	fatigue life correction factor
	T	=	fluid service temperature (°C)
	S*	=	S for $0 < $ sulfur content, $S \leq 0.015$ wt. %
		=	0.015 for $S > 0.015$ wt. %
	T*	=	0 for $T < 150°C$
		=	(T - 150) for $150 \leq T \leq 350°C$
	O*	=	0 for dissolved oxygen, DO < 0.05 parts per million (ppm)
		=	ln(DO/0.04) for 0.05 ppm $\leq DO \leq 0.5$ ppm
		=	ln(12.5) for DO > 0.5 ppm
	$\dot{\varepsilon}^*$	=	0 for strain rate, $\dot{\varepsilon} > 1\%$/sec
		=	$\ln(\dot{\varepsilon})$ for $0.001 \leq \dot{\varepsilon} \leq 1\%$/sec
		=	ln(0.001) for $\dot{\varepsilon} < 0.001\%$/sec

For Types 304 and 316 Stainless Steel : $F_{en} = \exp(0.935 - T^*\dot{\varepsilon}^*O^*)$

where:			
	F_{en}	=	fatigue life correction factor
	T	=	fluid service temperature (°C)
	T*	=	0 for T < 200°C
		=	1 for T ≥ 200°C
	$\dot{\varepsilon}^*$	=	0 for strain rate, $\dot{\varepsilon} > 0.4\%/\text{sec}$
		=	$\ln(\dot{\varepsilon}/0.4)$ for $0.0004 \le \dot{\varepsilon} \le 0.4\%/\text{sec}$
		=	$\ln(0.0004/0.4)$ for $\dot{\varepsilon} < 0.0004\%/\text{sec}$
	O*	=	0.260 for dissolved oxygen, DO < 0.05 ppm
		=	0.172 for DO ≥ 0.05 ppm

Based on the above, the bounding F_{en} multipliers for each material are as follows:

Low Alloy Steel:

For a PWR environment, DO < 0.05 ppm, and so O* = 0. Therefore, F_{en} is only dependent on temperature, as follows:

T (°C)	F_{en}
2	2.53
50	2.38
100	2.24
150	2.10
200	1.98
250	1.86
300	1.75

The bounding multiplier for low alloy steel is 2.53.
Carbon Steel:

For a PWR environment, DO < 0.05 ppm, and so O* = 0. Therefore, F_{en} is only dependent on temperature, as follows:

T (°C)	F_{en}
251	1.79
50	1.69
100	1.59
150	1.49
200	1.40
250	1.32
300	1.24

The bounding multiplier for carbon steel is 1.79.

Stainless Steel:

For a PWR environment, DO = 0, so O* = 0.260. T* = 0 for T < 200°C or T* = 1 for T > 200°C. Conservatively, T* is taken as 1. Therefore F_{en} is only dependent on the strain rate parameter $\dot{\varepsilon}$.

$\dot{\varepsilon} = 0$ for strain rate $\dot{\varepsilon} > 0.4\%/\text{sec}$, and $F_{en} = 2.55$
$\dot{\varepsilon} = \ln(\dot{\varepsilon}/0.4)$ for $0.0004 \le \dot{\varepsilon} \le 0.4\%/\text{sec}$, and $F_{en} = 2.55$ to 15.35
$\dot{\varepsilon} = \ln(0.0004/0.4)$ for $\dot{\varepsilon} < 0.0004\%/\text{sec}$, and $F_{en} = 15.35$

The bounding multiplier for stainless steel is 15.35.

Reactor Vessel Shell and Lower Head

The CUF calculated in the original Section III fatigue analysis for the reactor vessel shell and lower head was CUF = 0 (Reference 1). The environmental fatigue calculation was based on the worst-case Fen multiplier and is presented below (Reference 2):

Reactor Vessel Shell and Lower Head Region

Material: SA-336 Low Alloy Steel
Usage Factor (40 years): 0
Maximum Environmental Factor F_{en}: 2.53
Limiting Temperature: 0°C
Usage Factor (60 years): CUF_{60env} **= 0**

Reactor Vessel Inlet and Outlet Nozzles

The CUF calculated in the original Section III fatigue analysis for the reactor vessel inlet and outlet nozzles was CUF = 0.155 (Reference 1). The environmental fatigue calculation was based upon the worst case F_{en} multiplier and is presented below (Reference 2):

Reactor Vessel Inlet Nozzle

Material: SA 336 Low Alloy Steel
Usage Factor (40 years): 0.155
Limiting Temperature: 0°C
Maximum Environmental Factor F_{en}: 2.53

Usage Factor (60 years): CUF_{60env} **= 0.3922**

Reactor Vessel Outlet Nozzle

Material: SA 336 Low Alloy Steel
Usage Factor (40 years): 0.155
Limiting Temperature: 0°C
Maximum Environmental Factor F_{en}: 2.53

Usage Factor (60 years): CUF_{60env} **= 0.3922**

(b) The fatigue usage factors for the USAS B31.1 locations were calculated as follows:

Safety Injection-to-Cold Leg Branch Connection

An explicit fatigue analysis of the safety injection-to-cold leg RCS branch connection was performed according to the requirements of ASME Section III, Subsection NB-3600. All modes of operation and transient cases were evaluated. The results of this analysis are presented below (Reference 3):

Safety Injection-to-Cold Leg Branch Connection

Material: SA-182 Type 316 Stainless Steel
Usage Factor (40 years): 0.0164
Maximum Environmental Factor: 15.35

Usage Factor (60 years): CUF_{60env} = **0.2517**

RHR-to-Safety Injection Tee

An explicit fatigue analysis of the RHR-to-safety injection "tee" connection was performed according to the requirements of ASME Section III, Subsection NB-3600. All modes of operation and transient cases were considered. The results of this analysis are presented below (Reference 3):

RHR-to-Safety Injection Tee Connection

Material: SA-376 Type 316 Stainless Steel
Usage Factor (40 years): 0.0093
Maximum Environmental Factor: 15.35

Usage Factor (60 years): CUF_{60env} = **0.1428**

Charging Nozzles

The transient event that contributes to fatigue usage of the charging nozzles is loss of letdown flow with delayed return to service. An actual template set of real plant data from this transient event which occurred on January 7, 2003 was used to compute the incremental fatigue usage for the charging nozzles and the appropriate environmental factor (Reference 4).

Reactor Coolant Piping Charging Nozzles

Material: SA-376 Type 316 Stainless Steel
Usage Factor (per event): 0.00011
Environmental Factor: 7.56

Usage Factor (Environmental Effects): CUF_{env} = **0.000847**

The environmentally-assisted fatigue usage for the charging nozzles at Ginna Station will remain less than 1.0 for as many as 1181 events of loss of letdown flow with delayed return to service.

Pressurizer Lower Head and Surge Line

The EPRI FatiguePro software program was customized to monitor fatigue-critical locations in the surge line and pressurizer lower head in the Ginna plant. An analysis was performed based on available template sets of real plant data to determine the incremental fatigue usage factor for known plant transients, including the effects of "insurge/outsurge" and environmentally-assisted fatigue (EAF). Cumulative usage factors for the operating life of the plant were computed based on the results of real plant data, and expected future usage was computed using projections of expected plant cycles (see response to RAI 4.3.1-1).

The technical approach is summarized as follows:

- The flow rate in the surge line was computed based on a mass balance approach, using the incoming spray demand and the rate of change of the pressurizer water level, taking into account temperature effects.
- A 2-dimensional model was created to take into account (a) the advance and time delay of colder water from the hot leg into the surge line and lower head of the pressurizer, and (b) the heat transfer between the fluid and metal.
- This approach has been verified to be conservative based on available thermocouple data from another plant, as well as plant-specifically for Ginna Station by comparing the surge line temperature instrument reading with the FatiguePro-calculated water temperature in the region of the surge nozzle. The temperatures at the nozzle and lower head are calculated in FatiguePro completely independently from the surge line temperature instrument.
- Finite element models (including thermal sleeves in the pressurizer surge nozzle and hot leg RCS surge nozzle) were created to compute "Green's Function" stress responses to step changes in temperature at various zones in the pressurizer. Stresses could then be computed based on the calculated fluid temperatures at the various zones in the pressurizer and surge line.
- The stress history was used to compute fatigue usage in FatiguePro.

Significant temperature differentials (ΔTs, i.e., the difference between pressurizer water temperature and RCS hot leg temperature) are required to produce thermal fatigue in the surge line and lower head. These temperature differentials occur during plant heatup and cooldown cycles. Other transients such as plant trips do not produce stresses above the minimum fatigue threshold. Ginna Station uses the "water solid" method of heatup and cooldown, which maintains relatively small ΔTs during operation (typically less than 200°F) which results in a relatively benign effect on fatigue usage. Real plant data from various heatup/cooldown cycles since 1996 were analyzed to compute incremental fatigue usage for a heatup/cooldown cycle. The location with the highest fatigue usage in the pressurizer bottom head was determined to be the heater tube-to-lower head (penetration) weld. For the heater penetration location, the primary stress transient is not due to insurge and outsurge, but rather the general thermal expansion stress that arises from the global heatup and cooldown of the pressurizer. This location is a stainless steel weld to the tube and clad very close to the low alloy steel pressurizer shell. A high steady state dissimilar metal thermal expansion stress is established during the heatup and is relaxed during the cooldown. It is of a magnitude that overwhelms the small stress additions coming from insurges and outsurges of fluid. The next most fatigue sensitive location is the pressurizer surge nozzle. This location is affected most by insurges and outsurges, having essentially no steady state stress. This location has a much smaller stress concentration effect than the heater weld.

The cumulative usage factors for the heater penetration, pressurizer surge nozzle and surge line nozzle-to-RCS hot leg connection were calculated and the results are as follows (Reference 5):

Pressurizer Heater Penetration

Material: Type 316 Stainless Steel
Usage Factor (60 years[1]): 0.048
Maximum Environmental Factor: 15.35

Usage Factor (60 years): CUF_{60env} = **0.74**

Note 1: Fatigue usage factor was calculated based on 200 heatup and cooldown cycles

Pressurizer Surge Nozzle

Material: SA 376 Type 316 Stainless Steel
Usage Factor (60 years[1]): 6.276E-07
Maximum Environmental Factor: 15.35

Usage Factor (60 years): CUF_{60env} = **9.633E-06**

Note 1: Fatigue usage factor was calculated based on 200 heatup and cooldown cycles

RCS Hot Leg Surge Nozzle

Material: SA 376 Type 316 Stainless Steel
Usage Factor (60 years[1]): 0.0132
Maximum Environmental Factor: 15.35

Usage Factor (60 years): CUF_{60env} = **0.2022**

Note 1: Fatigue usage factor was calculated based on 200 heatup and cooldown cycles

Historical data from actual plant heatup and cooldown cycles from 1975 to 2002 was reviewed to more accurately account for early plant operation. During the early years of operation, the ΔTs during heatup and cooldown cycles were higher in some cases than those typically encountered in later years. A sensitivity analysis was performed by running simulated data with higher ΔTs (by lowering the hot leg temperature) in FatiguePro to establish a correlation between maximum ΔT and increase in fatigue usage factor. On the average, this resulted in approximately a 50% increase in incremental fatigue usage as compared with more recent plant operation. Assuming the full design set of 200 cycles, and assuming that the first 59 cycles of plant heatups and cooldowns occurred with the 50% increase in fatigue usage, the expected cumulative fatigue usage for the heater penetration location with the maximum environmental factor applied is expected to be 0.85.

FatiguePro will be used at Ginna Station to monitor future fatigue usage at all fatigue sensitive locations in the reactor coolant system.

References:

1. Babcock and Wilcox Stress Report, Contract No. 610-0110, "Summary Report", January 1971, SI File No. RGE-10Q-205

2. Structural Integrity Associates Calculation Package W-RGE-12Q-320

3. Structural Integrity Associates Calculation Package W-RGE-12Q-309

4. Structural Integrity Associates Calculation Package W-RGE-12Q-323

5. Structural Integrity Associates Calculation Package W-RGE-12Q-310

F-RAI 4.6 -1

Provide a list of design transients and corresponding cycles that were prescribed in the design of the containment penetrations.

Response

The Containment liner penetrations comply with ASME Code, Section III-1965 for pressure boundary and AISC Code for structural steel. Paragraph N-415.1 of ASME Section III-1965 states that a fatigue analysis is not required, and it may be assumed that the peak stress intensity limit has been satisfied for a vessel or component by compliance with the applicable requirements for materials, design, fabrication, testing and inspection. Provided the service loading of the vessel of component meets all of six conditions. The design transients for the six conditions and the allowable number of cycles are as follows:

Condition	Allowable Number of Cycles
1. Atmospheric to Operating Pressure Cycles	1500
2. Normal Service Pressure Fluctuations	17,145
3. Temperature Difference - Startup and Shutdown	120
4. Temperature Difference - Normal Service	17,145
5. Temperature Difference - Dissimilar Materials	10^6
6. Mechanical Loads	10

These design transients are defined in the response to RAI 4.6-2.

F-RAI 4.6 -2

For the penetration sleeve and the annular plate connecting the pressure piping to the sleeve, provide the analysis that shows that the six conditions of ASME Section III, Subsection A,

N-415.1, 1965, will be satisfied for the period of extended operation.

Response

ASME Code, Section III, N-415.1 states that a fatigue analysis is not required, and it may be assumed that the peak stress intensity limit has been satisfied for a vessel or component by compliance with the applicable requirements for materials, design, fabrication, testing, and inspection, provided the service loading of the vessel or component meets all of six conditions. Each of the six conditions is stated below, together with an analysis demonstrating compliance through the period of extended operation.

The pressure boundary components evaluated in the calculation include the liner adjacent to the penetration, the penetration sleeve, and the annular plate connecting the pressure piping to the sleeve. The liner and all penetration sleeves are made of carbon steel. Most of the annular plates are also carbon steel, and those that are not are made of stainless steel. Since the allowable alternating stress intensity, Sa, for stainless steel at any specific number of cycles is always greater than the allowable for carbon steel [(see ASME Code, Figures N-415(A) and (B)], the allowable stress intensity for carbon steel is used in all cases.

Condition 1: Atmospheric to Operating Pressure Cycles

> The specified number of times that the pressure will be cycled from atmospheric pressure to operating pressure and back to atmospheric pressure shall not exceed the number of cycles on the applicable fatigue curve corresponding to an Sa value of 3 times the Sm value for the material at operating temperature, where Sa is the allowable alternating stress amplitude, and Sm is the Design Stress Intensity.

UFSAR Table 5.1-4 estimates 200 heatup/cooldown full pressure cycles over a 40 year period. However, based on operating experience the total number of heatup/cooldown pressure cycles over a 60 year period is projected to be less than 120 (see response to RAI 4.3.1-1). For this evaluation, 120 full pressure cycles will be assumed.

The Design Stress Intensity at 100 degrees F for each of the materials from which the liner, sleeve and annular plate are constructed is listed below:

ASTM A-201 Gr B	Sm = 20.0 ksi
ASTM A-106 Gr B	Sm = 20.0 ksi
ASTM A-442 Gr 60	Sm = 20.0 ksi
ASTM A-516 Gr 70	Sm = 23.3 ksi*
ASTM A-240 Type 304	Sm = 20.0 ksi

*Based upon current code criteria

For Sa equal to 3Sm, or 70 ksi, the allowable number of cycles from Figure N-415(A) is 1,500, which exceeds the projected heatup/cooldown cycles of 120. Therefore, Condition 1 is satisfied.

Condition 2: Normal Service Pressure Fluctuations

> The specified full range of pressure fluctuations during normal operation shall not exceed the quantity (1/3) x Design Pressure x (Sa/Sm), where Sa is the value obtained from the applicable design fatigue curve for the total specified number of significant pressure fluctuations, and Sm is the Design Stress Intensity for the material at operating temperature.

The projected cycles for all other transients over a 60 year period are significantly less than the total cycles given in UFSAR Table 5.1-4 over a 40 year period (see response to RAI 4.3.1-1). The total number of design cycles from Table 5.1-4, excluding startup and shutdown, will be conservatively used for the 60 year period. This equals 14,500 (loading/unloading), plus 2,200 (load increase/decrease), plus 400 (reactor trip), plus 45 (test), or 17,145 cycles. For Sa equal to 30,000 psi, Sm equal to 23,300 psi, and a Design Pressure of 60 psi, the allowable number of full range pressure fluctuations equals 26 psi, which will bound any pressure fluctuations seen during normal operating conditions.

Therefore, Condition 2 is satisfied.

Condition 3: Temperature Difference – Startup and Shutdown

> The temperature difference in degrees F between any two adjacent points of the component during normal operation, and during startup and shutdown, does not exceed the quantity Sa/(2Eα), where Sa is the value obtained from the applicable design fatigue curve for the total specified number of significant startup/shutdown cycles, and E and α are the elastic modulus and coefficient of thermal expansion (instantaneous) at the mean value of temperatures at the two points.

The maximum temperature difference between any two points, and the distance between adjacent points, occurs at the main steam line penetration. The mean temperature of the insulated main steam pipe is 530 degrees F (UFSAR Section 3.2.2.1.5). Conservatively taking the containment ambient air temperature as 80 degrees F, which is the mean containment temperature above the operating floor minus one standard deviation, to be the temperature at the point where the annular plate meets the penetration sleeve, the maximum temperature difference is 450 degrees F.

Less than 120 full startup/shutdown cycles are projected over a 60 year period (see response to RAI 4.3.1-1). For 120 cycles, Sa = 180 ksi for carbon steel. Therefore:

Delta T = 180,000/2(7.12)(27.38) = 462 degrees F

where: 7.12×10^6 is the instantaneous coefficient of thermal expansion at 305 degrees F, and

27.38×10^6 is the elastic modulus at 305 degrees F

Since 462 > 450, Condition 3 is satisfied.

Condition 4: Temperature Difference – Normal Service

> The temperature difference in degrees F between any two adjacent points of the component does not change during normal operation by more than the quantity Sa/(2Eα), where Sa is the value obtained from the applicable design fatigue curve for the total specified number of significant temperature difference fluctuations.

During normal operation, the temperature in the main steam line fluctuates between 514 and 547 degrees F (UFSAR Section 3.2.2.1.5). Data from the Containment Temperature Monitoring Program shows that the standard deviation of the containment air temperature above the operating floor is less than 20 degrees F, which can be taken to be the temperature fluctuation at the edge of the annular plate. Conservatively assuming that these two temperature fluctuations occur out-of-phase results in a maximum fluctuation range of:

547 – 514 + 2 x 20 = 73 degrees F

The resulting temperature difference, assuming a total number of significant temperature difference fluctuations of 17,145, is:

$$\Delta T = 30{,}000/(2)(27.38)(7.12) = 77 \text{ degrees F}$$

where: Sa = 30,000 psi

Since 77>73, Condition 4 is satisfied.

Condition 5: Temperature Difference – Dissimilar Materials

> For components fabricated from materials of differing moduli of elasticity and/or coefficients of thermal expansion, the total range of temperature fluctuations experienced by the component during normal operation shall not exceed the magnitude Sa/[2(E1α1 – E2α2)], where Sa is the value obtained from the applicable design fatigue curve for the total specified number of significant temperature fluctuations. A temperature fluctuation shall be considered significant if its total excursion exceeds the quantity S/[2(E1α1 –
>
> E2α2)], where S is the value obtained from the applicable design fatigue curve for 10^6 cycles.

The only dissimilar material interface occurs at the junction of the carbon steel sleeve and stainless steel annular plate. The maximum difference between the products of E and α in the denominator occurs when considering a carbon steel sleeve welded to an austenitic stainless steel annular plate.

At

10^6 cycles, S = 13,000 psi. Thus, a significant temperature fluctuation must exceed:

13,000/[2(28.27 x 9.74 – 27.38 x 7.12)] = 81 degrees F

During normal operation, the temperature fluctuations at the junction of the penetration sleeve and annular plate are less than 81 degrees F. Therefore, Condition 5 is satisfied.

Condition 6: Mechanical Loads

> The specified full range of mechanical loads, excluding pressure, shall not result in load stresses whose range exceeds the Sa value obtained from the applicable design fatigue curve for the total specified number of significant load fluctuations.

Mechanical loads include dead load, pressure and seismic. Pressure loading is excluded and dead load is not cyclic. Therefore, the only mechanical load that must be considered is seismic. The number of maximum stress cycles considered during an SSE event is 10 (IEEE-344). At 10 cycles, Sa equals 550 ksi.

The maximum allowable stress intensity due to all loads (not just seismic) is 3Sm. Assuming a stress concentration factor of 5 (N-415.3), results in a maximum peak stress of 15Sm. The largest Sm value of all materials considered is 23.3 ksi, which results in a maximum peak stress of (15)(23.3) = 350 ksi. This is significantly less than the allowable alternating stress of 550 ksi. Therefore, Condition 6 is satisfied.

The fatigue evaluation for the liner and liner penetrations in accordance with the ASME Code, Section III, 1965 Edition requirements for Class B Vessels demonstrates that the liner and penetrations comply with the ASME Code, Section III, 1965 Edition requirements for fatigue through the period of extended operation.

F-RAI 4.7.3 -1

Provide the design transients and corresponding cycles which generated the static stress of 13,600 psi in the fillet weld attaching the channels to the liner.

Response

The fillet weld attaching the channel anchors to the liner was designed for 100,000 full stress cycles caused by fluctuations of temperature and pressure in the Containment.

F-RAI 4.7.3 -3

Provide justification why a fatigue-strength reduction factor was not applied to the stress caused by static loading for determining the allowable cycles for the fillet weld attaching the channel anchors to the liner.

Response

A fatigue analysis of the fillet weld attaching the channel anchors to the liner was performed as part of the original Containment design. The allowable fatigue stress of the attachment weld was set equal to the stress caused by static loading. This stress equals 13,600 psi and, based on the design codes referenced in the response to RAI 4.7.3-2, corresponds to 100,000 stress cycles. 100,000 cycles corresponds to more than four full stress cycles each day for 60 years of operation. Fluctuations of temperature and pressure in the Containment on a daily basis are not of sufficient magnitude to cause four full cycles of design basis stress at the liner anchorage weld every day. The fatigue analysis is therefore valid through the period of extended operation.

F-RAI 4.7.7 -1

The thermal aging embrittlement effect (loss of fracture toughness) on cast austenitic stainless steel is time dependent and is treated as a TLAA. The applicant performed a Leak-Before-Break (LBB/flaw tolerance) analysis to demonstrate that leaks from RCS piping can be detected prior to the cracks growing to a size that would become unstable. The applicant referenced a Westinghouse report (WCAP-15837, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the R. E. Ginna Nuclear power Plant for the License Renewal Program," April 2002) for its LBB analysis. The applicant also performed a fracture mechanics analysis in accordance with the requirements of ASME Code Case N-481 for the cast austenitic stainless steel (CF8M) reactor coolant pump (RCP) casings for the extended operation period. This fracture mechanics analysis was documented in a Westinghouse report (WCAP-15873, "A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Casings of R. E. Ginna Nuclear Power Plant for the License Renewal Program," April 2002). Code Case N-481 allows the required volumetric inspection of RCP casings to be replaced by a visual examination with the performance of an evaluation to demonstrate the safety and serviceability of the pump casings.

a) Confirm whether the two Westinghouse reports (WCAP-15837 and WCAP-15873) referenced in Section 4.7.7 have been submitted to NRC for review and approval. If these reports have been approved by NRC, identify the NRC approval documents. If these reports have not been reviewed and approved by NRC, submit the reports on the "docket" for Ginna's LRA.

b) If the reports have not been reviewed and approved by NRC, confirm whether the NRC approved methodologies including the material properties and other input parameters that were used in the analysis. Also identify areas in the referenced Westinghouse analyses that deviate from NRC recommended guidelines and provide justification for each deviation. If the requested information is already available in the referenced reports, summarize the information and identify the relevant sections in the reports.

Response

Two (2) copies each of Westinghouse Topical Reports WCAP-15873, "A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Casing of R. E. Ginna Nuclear Power Plant for the License Renewal Program", Proprietary Class 2, May 2002; WCAP-15837, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the R. E. Ginna Nuclear Power Plant for the License Renewal Program", Proprietary Class 2, April 2002; and the non-proprietary versions of these topical reports WCAP-15873-NP, Rev. 0, May 2003 and WCAP-15837-NP, Rev. 0, May 2003 were submitted under separate letter dated June 3, 2003.

F-RAI B2.1.7 -1

The Buried Piping and Tank Inspection Program consists of implementing preventive measure such as applying protective coating and periodic inspections, when inspection opportunities arise, to manage the corrosion effect on the external surfaces of buried carbon steel piping and tanks. In addition, the LRA states that this AMP is not specifically used for aging management at Ginna Station, as the inspection activities are performed through the One-Time Inspection Program.

a) Confirm and discuss whether this program is consistent with the guidelines provided in AMP XI.M34,"Buried Piping and Tanks Inspection" of NUREG-1801. Discuss all the deviations from AMP XI.M34 and provide justification for each deviation.

b) For each buried piping and tank, describe what preventive measures such as coating, wrapping or other protective measures are applied to mitigate the corrosion of its external surfaces. Confirm that the preventive measures applied are consistent with the guidance provided in NACE Standards RP-0285-95 and RP-0169-96.

c) Identify the environment that the inner surface of each buried piping and tank is exposed to and discuss its potential degradation caused by the environment. Also identify any scheduled maintenance that would provide the opportunity for inspection of the buried piping and tanks.

d) Discuss how the proposed inspection frequency based on the inspection of opportunity would provide adequate assurance that the corrosion of external surfaces of the buried piping and tanks will not occur when the opportunity for inspection does not arise.

e) The inspection activities of buried piping and tanks should be identified in the One-Time Inspection Program; if not, justify its exclusion.

f) Discuss the bases for not monitoring/inspecting the potential corrosion or degradation of the internal surfaces of the buried piping and tanks.

Response

As indicated in the response to RAI B2.1.8-1, the only buried tanks and piping within the scope of license renewal at Ginna Station are the emergency diesel generator (EDG) fuel oil storage tanks, the Technical Support Center (TSC) diesel fuel oil storage tank, fire-water piping, and sections of service water piping. The buried environment at Ginna Station is considered benign.

a) The Ginna Station Buried Piping and Tanks Inspection program is consistent with the guidelines provided in NUREG-1801, Section XI.M34.

b) See the response to RAI B2.1.8-1

c) The inner surfaces of the buried EDG diesel fuel oil storage tanks and the TSC diesel fuel oil storage tank are exposed to diesel fuel oil. As indicated in the response to RAI B2.1.8-1, an internal inspection of the underground EDG storage tanks is performed under the Periodic Surveillance and Preventive Maintenance (PSPM) program on a nine-year frequency. This activity includes cleaning, visual inspection and ultrasonic thickness measurements. All underground diesel fuel oil storage tanks are pressure-tested annually to verify leak-tightness. The interior of buried fire water piping is exposed to either service water (fresh Lake Ontario water) or city water. The interior of service water piping is exposed to service water (fresh Lake Ontario water). As discussed in the response to RAI 2.1.8-1, both external and internal inspections have been performed on buried fire water pipe and service water pipe during maintenance activities and the condition of the piping segments examined was found to be excellent.

d) As discussed in the response to RAI 2.1.8-1, several inspections of opportunity have verified that the external and internal condition of each type of buried piping and tanks at Ginna Station is excellent. Plant-specific operating experience over the past 33 years, therefore, indicates that future inspections of opportunity will provide adequate assurance that corrosion of external surfaces of buried piping and tanks will be managed so that the intended function of the buried components will be maintained during the period of extended operation.

e) Inspections of buried piping and tanks is now included in the One-Time Inspection Program.

f) As discussed in the response to (c) above and RAI 2.1.8-1, periodic inspections of the interior surfaces of the underground EDG fuel oil storage tanks are performed under the PSPM program. The results of inspections of opportunity of buried piping performed to date indicate that monitoring of internal surfaces of buried fire water and service water piping is not necessary. Future inspections of opportunity will provide adequate assurance that corrosion of internal surfaces of buried piping will be managed so that the intended function of the buried components will be maintained during the period of extended operation.

<u>D-RAI B2.1.15 -1</u>

In order for the staff to evaluate the acceptability of the Flow Accelerated Corrosion (FAC) Program, the applicant is requested to provide a list of the components in the program most susceptible to FAC. The list should include initial wall thickness (nominal), current wall thickness and the future predicted wall thickness.

<u>Response</u>

Two examples of sections of the secondary system within the scope of the Ginna Station Erosion-Corrosion Program (which implements the Flow-Accelerated Corrosion Program) are provided as follows (details of data is provided in Attachment 3) :

Component Equipment Identification Number (EIN): M21-39A
Steam Extraction to Preseparator and 4B Low Pressure Heater.
16" Sch 40S - A234/WPB/WPB
Tnom = .375in
Tmeasured –RFO2002 = .212in
Tnom-baseline-RFO2002 = will be replaced per Work Order # WO20200859 in RFO 2003
Average Wear Rate per CHECWORKS is 8.8mils/year
Current Wear Rate per CHECWORKS is: 6.9mils/year
Predicted minimum wall thickness in RFO2003 is:
.212in-(.0088in/year)(1.5year) = .199in
Calculated minimum wall is: .199in due to Tnom /2 criteria *.

Component Equipment Identification Number (EIN): M75-29A
Steam Extraction to 5B High Pressure Heater.
12" Sch40 – A106/B/B
Tnom = .375in
Tmeasured –RFO1999 = .304in

Average Wear Rate per CHECWORKS is 18.1mils/year
Current Wear Rate per CHECWORKS is: 10.1mils/year
Predicted minimum wall thickness in RFO2003 is:
.304in-(.0101in/year)(1.5year) = .289in
Calculated minimum wall is: .271in due to Dead weight + Longitudinal Pressure Stress criteria *.

* At GINNA, piping components susceptible to FAC are compare against to the maximum of the following calculated thickness values.
1- Thickness due to Tnom/2
2- Thickness due to Hoop Pressure
3- Thickness due to Dead Weight + Longitudinal Pressure.

Additional examples , if desired, can be viewed on site during subsequent inspections.

<u>D-RAI B2.1.15 -2</u>

The FAC Program at Ginna includes a prediction of the wall thinning for the components susceptible FAC. The wall thinning is predicted by the EPRI's CHECWORKS computer code. In order to allow the staff to evaluate the accuracy of these predictions, the applicant is requested to provide a few examples of the components for which wall thinning is predicted by the code and at the same time measured by UT or any other method employed in the applicant's plant.

<u>Response</u>

The requested information has been provided in response to F-RAI B2.1.15-1.

<u>F-RAI B2.1.23 -3</u>

The Periodic Surveillance and Preventative Maintenance program is an existing program that covers a wide range of systems, structures, and components. The LRA states that the program includes periodic replacement or refurbishment of equipment based on operating experience. It is not clear whether equipment in scope of LR is subject to periodic replacement or refurbishment, or whether the equipment can perform its intended function at the time it is replaced or refurbished. Clarify whether any equipment that requires aging management per 10 CFR Part 54 is managed by periodic replacement or refurbishment, whether any inspections are performed in addition to the periodic replacement or refurbishment, the basis for the replacement or refurbishment period, and the equipment operating experience.

<u>Response</u>

Roughing filters, containment isolation flange o-rings, radiation monitoring vacuum pumps, and auxiliary feedwater pump lube oil coolers are all subject to periodic replacement. Inspections performed on the equipment after it is removed from service are a critical inputs in establishing replacement frequencies. The basis for replacement frequencies is thus established through a combination of plant specific operating experience, industry operating experience and vendor recommendations. The specific equipment affected can perform its intended function at the time of replacement or refurbishment.

Other component types within the scope of license renewal such as pumps, valves, heat exchangers, etc. also subject to the Periodic Surveillance and Preventive Maintenance (PSPM) program receive refurbishment at set intervals which are also established as set forth above.

If a component in scope to license renewal was already included in the PSPM program, the PSPM program was credited for aging management. If an established PSPM activity required enhancement to satisfy the aging management requirements, a tracking mechanism was put in place to revise specific instructions in appropriate implementing procedures to include all necessary inspections for all applicable aging effects for each PSPM program activity. Based on the results of these aging management activities, inspection frequencies may be adjusted.

Attachment 2

RAI 2.1-4 response supplementary information

As a result of the scoping and screening and aging management reviews associated with RAI 2.1-4 the following changes are in effect:
In the license renewal application section 2.3.3.7, Heating Steam system description, 2nd paragraph, last sentence, revise: "As a result of these analyses and modifications, the only portion of the Heating Steam system considered as non-safety components whose failure could prevent the accomplishment of a safety function are those portions of the system contained in the Diesel Generator rooms" to read "As a result of these analyses and modifications, the only portion of the Heating Steam system considered as non-safety components whose failure could prevent the accomplishment of a safety function includes portions of the system located in the Diesel Generator rooms and Screen House. In the Screen House, components selected for boundary inclusion were evaluated based on the potential effects from boiler explosions, fuel fires, and steam releases and steam jet effects in proximity to safety related equipment".

In Table 2.3.3-7 add:

Component Group	Passive Function	Aging Management Reference
Boiler Package	Pressure Boundary	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (473)
Pipe	Pressure Boundary	Table 3.4-2 Line Number (474)
Valve Body	Pressure Boundary	Table 3.4-1 Line Number (5) Table 3.4-2 Line Number (386) Table 3.4-2 Line Number (425) Table 3.4-2 Line Number (429)

Change Basis: The aging management review boundary now includes the house heating boiler, the boiler steam main piping (until it exits the building underground), the boiler safety relief valves, and the gas fuel supply from where it enters the building from underground to the boiler. Additionally, local area heaters, pipe, traps housing and strainer housing located near vital electric busses 17 and 18, and the heater, piping, trap housing and strainer housings near the motor driven fire pump as well as the basement heater in the vicinity of vital cables are included. Affected components are shown on drawing 33013-1917. The above changes adds the carbon steel valve bodies, boiler pressure boundary, and piping associated with the heating steam system (treated water secondary >120 degrees F) located in the Screen House. Additionally they account for the carbon steel pipe and valve bodies and bronze valve bodies associated with the natural gas fuel (air and gas) used for the boiler and located in the Screen House.

In Table 3.4-2 add:

Component Type	Material	Environment	AERMs	Program /Activity	Discussion
(473) Boiler Package	Carbon Steel	Treated Water Secondary >120F	Loss of Material	Water Chemistry Control Program/ Preventive Maintenance and Periodic Surveillance 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.

RAI 2.3.3.13-3 (Clarification)

A typographical error was included in this RAI response, PT-111 should have been identified in response a.), rather an PT-112.

RAI 4.3.5-1 (Clarification)

The initial response to RAI 4.3.5-1 stated that ultrasonic examinations performed in 1999 characterized one indication (N2B-1) as a grouping of slag inclusions which was sized using 15° focused beam search units. This grouping of inclusions had been previously dispositioned as unacceptable. However, a small indication in the grouping which aligned in the through-wall direction only and did not affect the length measurement was determined to be only 34% DAC (Distance Amplitude Correction) and therefore did not need to be considered when determining the total flaw dimension. Based on this analysis, indication N2B-1 was determined as acceptable according to Section XI acceptance standard IWB-3512-1.

In 1999, Ginna Station was committed to the 1986 Edition (No Addenda) of the ASME Boiler and Pressure Vessel Code. Section XI of the 1986 Code Edition (No Addenda), Paragraph IWA-2232 (a) "Ultrasonic Examination" directs that ultrasonic examination of vessel welds in ferritic materials greater than 2 in. in thickness shall be conducted in accordance with Article 4 of ASME Section V. Paragraph T441.3.2.8 of Article 4 of Section 5 (1986 Edition, No Addenda) defines a recordable indication as a reflector which produces a response equal to or greater than 50% of DAC. Therefore, indications less than 50% of DAC do not require evaluation and indication N2B-1 was determined to be acceptable.

Starting in 2000, Ginna Station is committed to the 1995 Edition (1996 Addenda) of the ASME Boiler and Pressure Vessel Code. Paragraph T441.3.2.8 of Article 4 of Section V (1995 Edition, 1996 Addenda) now defines a recordable indication as a reflector which produces a response equal to or greater than 20% of DAC. Therefore, it is possible that the small indication in the grouping of slag inclusions may produce a response greater than 20% of DAC during the next vessel weld examination in 2009 and indication N2B-1 may again be dispositioned as unacceptable.

However, fracture mechanics analyses were performed by Teledyne Engineering Services in 1979 and by Structural Integrity Associates in 1989 to evaluate the stability and structural significance of flaw N2B-1. These analyses were required since the indication had been dispositioned as unacceptable in both 1979 and 1989. These analyses were both submitted to the NRC along with other material related to sizing of the indication on May 4, 1989. These analyses concluded that the only stresses of significance acting across the flaw are those due to vessel pressurization and weld residual stress, and concluded that the flaw satisfied the ASME Section XI Code criteria for acceptance by evaluation.

Additional conclusions in the Structural Integrity analysis (which is included as an attachment to this response) were as follows:

- Irradiation effects from the core are negligible at the flaw location;
- The applied stress intensity (K) for the embedded flaw with a through-wall dimension of 0.48 inches and a length of 4.94 inches is calculated as 7351 psiÖin due to pressure loading and weld residual stress;
- The above K value provides a margin of 27.2 against an upper shelf reference K (K_{IR}) of 200,000 psiÖin, compared to the required Section XI margin of 3.16; and
- Predicted fatigue crack growth, even for 1200 full cycles of vessel pressurization, is insignificant.

It has been determined that the number of design basis transients for Ginna Station for 40 years remains bounding for the period of extended operation (see response to RAI 4.3.1-1). The number of design cycles for heatups and cooldowns, and therefore vessel pressurizations, is 200. It is therefore concluded that, since irradiation effects at the flaw location are negligible and fatigue crack growth is insignificant even for 1200 cycles of pressurization, the flaw will remain stable and of no structural significance for the period of extended operation.

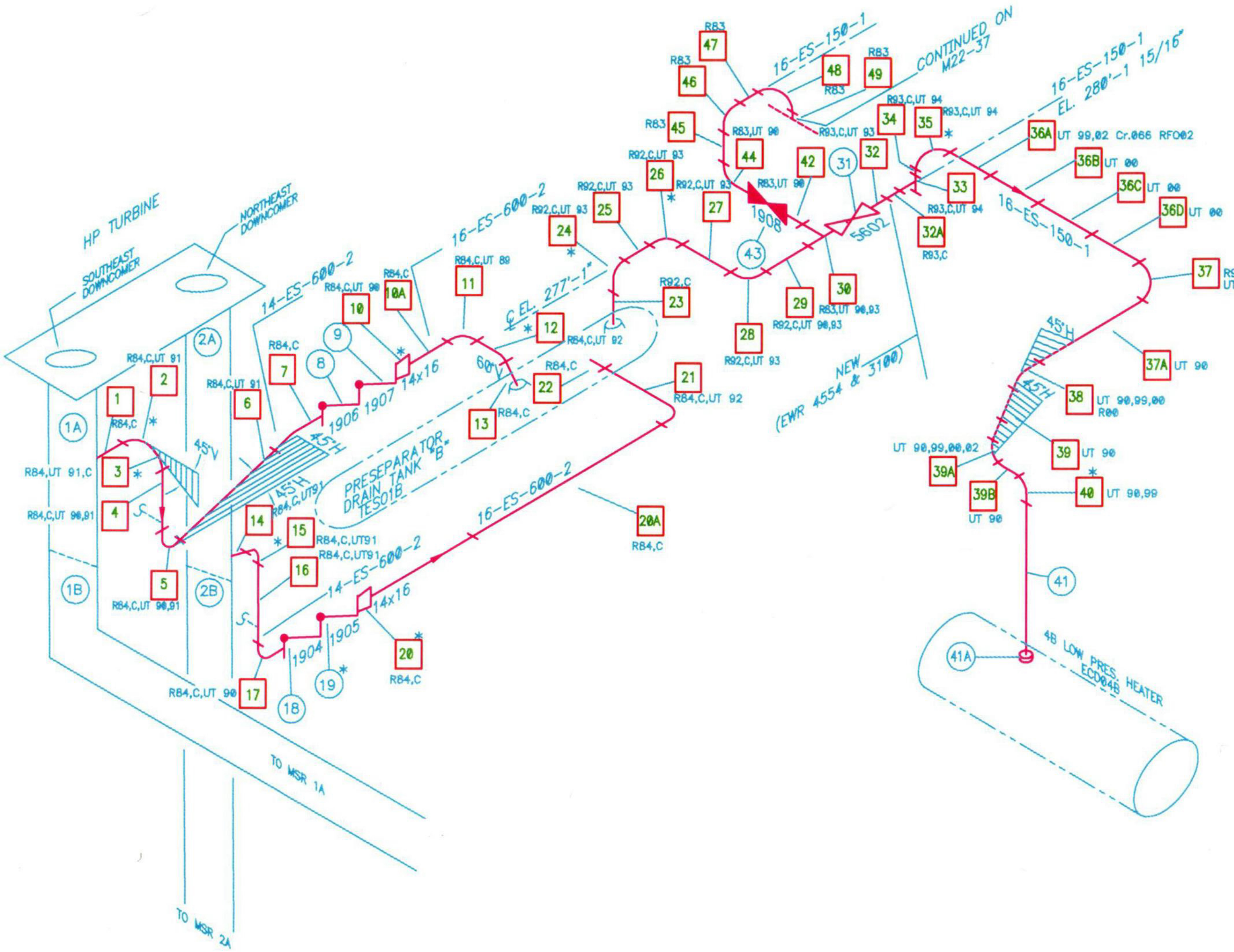
<u>RAI B2.1.16-1 (Clarification)</u>

NUREG-1801 refers to several ASTM Standards: D4057 for guidance on oil sampling, D1796 and D2709 for determination of water and sediment contaminants in diesel fuel, and modified D2276 Method A for determination of particulates. The methodology of D4057 is used at Ginna

Station for guidance on oil sampling. Either D1796 or D2709 may be used for determination of water and sediment content in fuel oil samples; D1796 requires that a solvent be added to the sample, whereas D2709 does not. Both methods provide results as percent of total contaminants. D975-78 specifies the method described in D1796 for water and sediment determination.

Tests in accordance with ASTM D4176 "Free Water and Particulate Contamination in Distillate Fuels (Clear and Bright Pass/Fail Procedures)" are also being performed as required by Ginna Station Technical Specifications. However, since the fuel may contain a red dye, the qualitative "Clear and Bright" criterion may be difficult to measure, such that the presence of free water or particulate could be obscured. RG&E is therefore in the process of initiating a request to the NRC for a change to the Technical Specifications to incorporate Industry/TSTF Standard Technical Specification Change Traveler TSTF-374. This traveler provides for the option of using the D1796 or D2709 tests for new fuel in lieu of the D4176 "clear and bright" test. Tests performed in accordance with D1796 or D2709 are acceptable methods for determination of water and sediment content. In addition, determination of particulates will be performed in accordance with ASTM standards (D2276 or its successor). The elimination of the D4176 "Clear and Bright" test and the addition of the alternative particulate test will take place when Technical Specifications are changed.

ATTACHMENT 3
RESPONSE FOR RAIs B2.1.15-1 AND B2-1.15-2



REF. DWG.S
PIPING DWG. 33013-1368
PIPING DWG. 33013-1369
PIPING DWG. 33013-1373
PIPING DWG. D-304-041
PIPING DWG. D-304-042
P&ID 33013-1903
WALKDOWN SKETCH (RG&E LETTER 13N1-RO-L0324 DATED 5/4/90)

NORTH

○ — DENOTES EROSION CORROSION COMPONENT I.D.
□ — DENOTES COMPONENT WITH CHECWORKS HISTORY
* — DENOTES FITTING WHICH HAS ADJACENT FITTING LOCATED WITHIN 1 DIAMETER UPSTREAM
UT — ULTRASONIC TESTING
RT — RADIOGRAPHY TESTING
R — DENOTES RFO WHEN COMPONENT WAS REPLACED
C — DENOTES COMPONENT AS CHROME-MOLY
B — DENOTES BASELINE UT INSPECTION

— DRAWING TO BE USED FOR COMPONENT IDENTIFICATION PURPOSES ONLY

CHECWORKS LINE/RUN NAME:
PRE-SEP TANK HTR 4 (1ES-5)/PRESEP TANK TO HTR 4
PRE-SEP TO TANK M21 (1ES-4)/EXT. STEAM 4th POINT

NUMBER	REVISION	DRAWN BY	CHECKED	RESP. ENG.	REVIEWER
3	MARKING CHECWORKS COMPONENT HISTORY	AJV3 9/30/02			
2	AS-BUILT	AJV3 7/8/97	AB 7/8/97	AB 7/8/97	MJS 7/8/97

DATE	RELEASED FOR	ENGR.
	CONSTRUCTION	
	LIMITED CONSTRUCTION: AS NOTED	
	PRELIMINARY NOT FOR CONSTRUCTION	
	BIDDING PURPOSES	
	INSPECTION	

Ginna Station (FACILITY) — 1M0511 (CAD FILENAME) — ROCHESTER GAS & ELECTRIC CORP. ROCHESTER, NEW YORK
SCALE NONE

EROSION CORROSION ISI PROGRAM
ISOMETRIC
STEAM EXTRACTION TO PRESEP. TANK B AND 4B LP HEATER

JOB NO.	FIGURE	DRAWING NO.	REV.
EWR 4499	M-21	10904-511	3

C01

Company : Rochester Gas and Electric
Plant : R. E. Ginna
Unit : 0
DB Name : GINNA

Report Date : 27-MAY-03 Time : 14:04:20

CHECWORKS FAC Version 1.0G (Build 75)

****
*** UT Summary ***
****

Component : M75-29A
Line : 5TH POINT ES (1ES-1)
Geometry Type : STRAIGHT PIPE

===> Section : U/S Main
Tnom = 0.375 (in), Tinit = 0.375 (in), Tscreen = 0.215 (in)

Period(s)	Grid Size (RxC)	No. of Points	Avg. Thk.	Standard Dev.	Min. Thk. (RxC)	Max. Thk. (RxC)	Total Life Wear	Wear Method	Total Svr. Hours.
RFO 1993	4x14	51	0.708	0.028	0.642(2,G)	0.771(1,A)	0.000	User-specified	159681.0
RFO 1999	3x14	42	0.360	0.028	0.304(3,B)	0.398(2,I)	0.090	Band	207721.0

===> Section : U/S Ext.
Tnom = 0.000 (in), Tinit = 0.375 (in), Tscreen = 0.000 (in)

Period(s)	Grid Size (RxC)	No. of Points	Avg. Thk.	Standard Dev.	Min. Thk. (RxC)	Max. Thk. (RxC)	Total Life Wear	Wear Method	Total Svr. Hours.
RFO 1993	4x14	56	0.655	0.093	0.430(1,E)	0.753(3,M)	0.000	User-specified	----

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 14:00:25
Analysis Date: 02-DEC-2002 Time: 10:09:50
CHECWORKS FAC Version 1.0G (Build 75)

****
*** Wear Rate Analysis: Wear Rates/Input Data Report ***
****

Run Name: Ext. Steam 5th Point
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.268

Duty Factor (Global): 1.000
Exclude Measure Wear: No

===>Grouped by Line: 5TH POINT ES (1ES-1), No Sorting.

Component Name	Geom. Code	Average Wear Rate (mils/year)	Current Wear Rate (mils/year)	Temp. (F)	Velocity (ft/s)	Steam Quality	Diameter (in)
M75-01A(L/E)	18	0.045	0.033	435.6	56.525	0.941	12.750
M75-01A(S/E)	18	0.056	0.041	435.6	74.079	0.941	10.750
M75-01	68	2.876	1.571	435.6	56.525	0.941	12.750
M75-02(U/S)	13	13.199	7.209	435.6	39.670	0.941	12.750
M75-02(D/S)	13	13.199	7.209	435.6	39.670	0.941	12.750
M75-02(BR.)	13	0.000	0.000	0.0	0.000	0.000	12.750
M75-03A	63	2.265	1.237	435.6	39.670	0.941	12.750
M75-03B	9	2.768	1.512	435.6	39.670	0.941	12.750
M75-03C	9	2.768	1.512	435.6	39.670	0.941	12.750
M75-03D	9	2.768	1.512	435.6	39.670	0.941	12.750
M75-03E	9	2.768	1.512	435.6	39.670	0.941	12.750
M75-04(U/S)	15	0.057	0.049	435.6	39.670	0.941	12.750
M75-04(D/S)	15	0.050	0.043	435.6	39.670	0.941	12.750
M75-05-5603	22	10.789	5.893	435.6	63.476	0.941	12.750
M75-05A	58	4.316	2.357	435.6	63.476	0.941	12.750
M75-06-5515	25	11.769	6.428	435.6	63.476	0.941	12.750
M75-06A	58	0.043	0.037	435.6	63.476	0.941	12.750
M75-07-5517	25	11.769	6.428	435.6	63.476	0.941	12.750
M75-08	58	4.316	2.357	435.6	63.476	0.941	12.750
M75-09(U/S)	12	9.704	6.755	435.6	39.670	0.941	12.750
M75-09(D/S)	12	9.704	6.755	435.6	39.670	0.941	12.750
M75-09(BR.)	12	0.000	0.000	0.0	0.000	0.000	12.750
M75-10	62	1.765	1.229	435.6	40.019	0.941	12.750
M75-11	4	5.450	3.794	435.6	39.670	0.941	12.750
M75-12A	54	5.644	3.082	435.6	39.670	0.941	12.750
M75-12B	9	2.768	1.512	435.6	39.670	0.941	12.750
M75-12C	9	2.768	1.512	435.6	39.670	0.941	12.750
M75-13	2	5.450	3.794	435.6	39.670	0.941	12.750
M75-14	2	1.580	1.100	435.6	39.670	0.941	12.750
M75-15A	52	0.169	0.167	435.6	41.038	0.853	12.750
M75-15	9	14.822	9.612	435.6	63.213	0.782	12.750
M75-15B	9	1.366	0.999	435.6	56.525	0.941	12.750
M75-16A(L/E)	18	2.834	2.074	435.6	56.525	0.941	12.750
M75-16A(S/E)	18	3.573	2.615	435.6	74.079	0.941	10.750
M75-16	68	2.876	1.571	435.6	56.525	0.941	12.750
M75-17(U/S)	13	13.199	7.209	435.6	39.670	0.941	12.750
M75-17(D/S)	13	13.199	7.209	435.6	39.670	0.941	12.750
M75-17(BR.)	13	0.000	0.000	0.0	0.000	0.000	12.750
M75-18A	63	2.265	1.237	435.6	39.670	0.941	12.750
M75-18B	9	2.768	1.512	435.6	39.670	0.941	12.750
M75-18C	9	2.157	1.178	435.6	39.670	0.941	12.750
M75-18D	9	2.768	1.512	435.6	39.670	0.941	12.750
M75-30(U/S)	15	5.659	3.090	435.6	39.670	0.941	12.750
M75-30(D/S)	15	4.981	2.720	435.6	39.670	0.941	12.750
M75-19-5604	22	10.789	5.893	435.6	63.476	0.941	12.750
M75-19A	58	4.316	2.357	435.6	63.476	0.941	12.750
M75-20-5514	25	11.769	6.428	435.6	63.476	0.941	12.750
M75-20A	58	0.037	0.037	435.6	63.476	0.941	12.750
M75-21-5516	25	11.769	6.428	435.6	63.476	0.941	12.750
M75-22	58	3.297	2.357	435.6	63.476	0.941	12.750
M75-23(U/S)	12	12.368	6.755	435.6	39.670	0.941	12.750
M75-23(D/S)	12	12.368	6.755	435.6	39.670	0.941	12.750
M75-23(BR.)	12	0.000	0.000	0.0	0.000	0.000	12.750
M75-23A	62	1.719	1.229	435.6	40.019	0.941	12.750
M75-24	4	5.307	3.794	435.6	39.670	0.941	12.750
M75-25	54	7.242	3.955	435.6	39.670	0.941	12.750
M75-26	2	5.450	3.794	435.6	39.670	0.941	12.750
M75-27	52	3.700	2.575	435.6	39.670	0.941	12.750
M75-28	2	5.450	3.794	435.6	39.670	0.941	12.750
M75-29A	52	17.797	9.939	435.6	36.687	0.853	12.750
M75-29	9	16.913	9.612	435.6	63.213	0.782	12.750
M75-29B	9	15.053	8.536	435.6	38.102	0.782	12.750

===>Grouped by Line: 5TH PT ES CROSS "T" (2ES-1), No Sorting.

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 14:02:19
Analysis Date: 02-DEC-2002 Time: 10:09:50
CHECWORKS FAC Version 1.0G (Build 75)

****
*** Wear Rate Analysis: Inspection History Report ***
****

Run Name: Ext. Steam 5th Point
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.268

Duty Factor (Global): 1.000
Exclude Measure Wear: No

===>Grouped by Line: 5TH POINT ES (1ES-1), No Sorting.

Component Name	Geom. Code	Material No.	Material Cr. (%)	Material Cu. (%)	Material Mo. (%)	Sigma (psi)	Time (hrs) Last Inspected	Time (hrs) Replaced	Analysis Option	Measured Wear (mils)
M75-01A(L/E)	18	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	18	21	0.00	0.00	0.00	15000	137574	137574		41
M75-01A(S/E)	18	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	18	21	0.00	0.00	0.00	15000	137574	137574		157
M75-01	68	5	0.00	0.00	0.00	15000	166985	-----		67
M75-02(U/S)	13	21	0.00	0.00	0.00	15000	-----	-----		---
M75-02(D/S)	13	21	0.00	0.00	0.00	15000	-----	-----		---
M75-02(BR.)	13	21	0.00	0.00	0.00	15000	-----	-----		---
M75-03A	63	5	0.00	0.00	0.00	15000	220721	-----		54
M75-03B	9	5	0.00	0.00	0.00	15000	220721	-----		78
M75-03C	9	5	0.00	0.00	0.00	15000	-----	-----		---
M75-03D	9	5	0.00	0.00	0.00	15000	220721	-----		61
M75-03E	9	5	0.00	0.00	0.00	15000	-----	-----		---
M75-04(U/S)	15	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	15	21	0.00	0.00	0.00	15000	-----	166985		---
M75-04(D/S)	15	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	15	21	0.00	0.00	0.00	15000	-----	166985		---
M75-05-5603	22	93	0.00	0.00	0.00	14000	-----	-----		---
M75-05A	58	5	0.00	0.00	0.00	15000	166985	-----		78
M75-06-5515	25	93	0.00	0.00	0.00	14000	-----	-----		---
M75-06A	58	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	58	5	0.00	0.00	0.00	15000	166985	166985		186
M75-07-5517	25	93	0.00	0.00	0.00	14000	-----	-----		---
M75-08	58	5	0.00	0.00	0.00	15000	182553	-----		94
M75-09(U/S)	12	21	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	12	21	0.00	0.00	0.00	15000	-----	121957		---
M75-09(D/S)	12	21	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	12	21	0.00	0.00	0.00	15000	-----	121957		---
M75-09(BR.)	12	21	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	12	21	0.00	0.00	0.00	15000	-----	121957		---
M75-10	62	5	0.00	0.00	0.00	15000	220721	-----		94
*Replacement #1	62	5	0.00	0.00	0.00	15000	-----	121957	Excl LCF	---
M75-11	4	21	0.00	0.00	0.00	15000	220721	-----		58
*Replacement #1	4	21	0.00	0.00	0.00	15000	-----	121957	Excl LCF	---
M75-12A	54	5	0.05	0.00	0.00	15000	233721	-----		151
M75-12B	9	5	0.00	0.00	0.00	15000	220721	-----		119
M75-12C	9	5	0.00	0.00	0.00	15000	-----	-----		---
M75-13	2	21	0.00	0.00	0.00	15000	233721	-----		69
*Replacement #1	2	21	0.00	0.00	0.00	15000	-----	121957	Excl LCF	---
M75-14	2	21	0.15	0.00	0.00	15000	233721	-----		41
*Replacement #1	4	21	0.00	0.00	0.00	15000	-----	121957	Excl LCF	---
M75-15A	52	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	12	5	0.00	0.00	0.00	15000	-----	207721		---
M75-15	9	5	0.00	0.00	0.00	15000	207721	-----		171
M75-15B	9	5	0.05	0.00	0.00	15000	233721	-----		53
M75-16A(L/E)	18	21	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	18	21	0.00	0.00	0.00	15000	137574	137574		45
M75-16A(S/E)	18	21	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	18	21	0.00	0.00	0.00	15000	137574	137574		104
M75-16	68	5	0.00	0.00	0.00	15000	152457	-----		73
M75-17(U/S)	13	21	0.00	0.00	0.00	15000	-----	-----		---
M75-17(D/S)	13	21	0.00	0.00	0.00	15000	-----	-----		---
M75-17(BR.)	13	21	0.00	0.00	0.00	15000	-----	-----		---
M75-18A	63	5	0.00	0.00	0.00	15000	-----	-----		---
M75-18B	9	5	0.00	0.00	0.00	15000	-----	-----		---
M75-18C	9	5	0.05	0.00	0.00	15000	233721	-----		58
M75-18D	9	5	0.00	0.00	0.00	15000	-----	-----		---
M75-30(U/S)	15	21	0.00	0.00	0.00	15000	-----	-----		---
M75-30(D/S)	15	21	0.00	0.00	0.00	15000	-----	-----		---
M75-19-5604	22	93	0.00	0.00	0.00	14000	-----	-----		---
M75-19A	58	5	0.00	0.00	0.00	15000	-----	-----		---
M75-20-5514	25	93	0.00	0.00	0.00	14000	-----	-----		---
M75-20A	58	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	58	5	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #2	58	26	1.90	0.00	0.87	15000	-----	233721		---
M75-21-5516	25	93	0.00	0.00	0.00	14000	-----	-----		---
M75-22	58	5	0.00	0.00	0.00	15000	182553	-----		51
*Replacement #1	58	5	0.00	0.00	0.00	15000	130865	130865		113
M75-23(U/S)	12	21	0.00	0.00	0.00	15000	-----	-----		---
M75-23(D/S)	12	21	0.00	0.00	0.00	15000	-----	-----		---
M75-23(BR.)	12	21	0.00	0.00	0.00	15000	-----	-----		---
M75-23A	62	5	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	62	5	0.00	0.00	0.00	15000	-----	130865		---
M75-24	4	21	0.00	0.00	0.00	15000	166985	-----		98
*Replacement #1	4	21	0.00	0.00	0.00	15000	130865	130865		78
M75-25	54	5	0.00	0.00	0.00	15000	-----	-----		---
M75-26	2	21	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	2	21	0.00	0.00	0.00	15000	-----	121957		---
M75-27	52	5	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	52	5	0.00	0.00	0.00	15000	-----	121957		---
M75-28	2	21	0.00	0.00	0.00	15000	207721	-----		50
*Replacement #1	2	21	0.00	0.00	0.00	15000	-----	121957	Excl LCF	---

M75-29A	52	5	0.00	0.00	0.00	15000	-----	-----	Excl LCF	---
M75-29	9	5	0.00	0.00	0.00	15000	-----	-----	Excl LCF	---
M75-29B	9	5	0.00	0.00	0.00	15000	-----	-----	Excl LCF	---

===>Grouped by Line: 5TH PT ES CROSS "T" (2ES-1), No Sorting.

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 14:02:26
Analysis Date: 02-DEC-2002 Time: 10:09:50
CHECWORKS FAC Version 1.0G (Build 75)

*** Wear Rate Analysis: Thickness/Service Time Report ***

Run Name: Ext. Steam 5th Point
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.268

Duty Factor (Global): 1.000
Exclude Measure Wear: No

Component Name	Thickness (in) Init.	Prd.[1]	Thoop	Tcrit	Component Predicted[1] Time to Tcrit (hrs) Non-Inspected	Inspected	Component Actual Service Time (hrs)
===>Grouped by Line: 5TH POINT ES (1ES-1), No Sorting.							
M75-01A(L/E)	0.375	0.398	0.179	0.215	48918472	------	122147
M75-01A(S/E)	0.375	0.375	0.151	0.198	37717044	------	122147
M75-01	0.375	0.318	0.179	0.215	------	572545	259721
M75-02(U/S)	0.375	0.250	0.179	0.215	42051	------	259721
M75-02(D/S)	0.375	0.285	0.179	0.215	84583	------	259721
M75-02(BR.)	0.375	0.281	0.179	0.215	99000000	------	259721
M75-03A	0.375	0.349	0.179	0.215	------	951585	259721
M75-03B	0.375	0.318	0.179	0.215	------	597738	259721
M75-03C	0.375	0.293	0.179	0.215	451648	------	259721
M75-03D	0.375	0.349	0.179	0.215	------	777385	259721
M75-03E	0.375	0.293	0.179	0.215	451648	------	259721
M75-04(U/S)	0.375	0.403	0.179	0.215	33872132	------	92736
M75-04(D/S)	0.375	0.417	0.179	0.215	41359316	------	92736
M75-05-5603	0.375	0.055	0.191	0.215	-156148	------	259721
M75-05A	0.375	0.275	0.179	0.215	------	223128	259721
M75-06-5515	0.375	0.026	0.191	0.215	-164741	------	259721
M75-06A	0.375	0.377	0.179	0.215	38075700	------	92736
M75-07-5517	0.375	0.026	0.191	0.215	-164741	------	259721
M75-08	0.375	0.314	0.179	0.215	------	367697	259721
M75-09(U/S)	0.375	0.609	0.179	0.215	511441	------	137764
M75-09(D/S)	0.375	0.467	0.179	0.215	327289	------	137764
M75-09(BR.)	0.375	0.371	0.179	0.215	99000000	------	137764
M75-10	0.375	0.275	0.179	0.215	------	430811	137764
M75-11	0.375	0.341	0.179	0.215	------	290485	137764
M75-12A	0.375	0.215	0.179	0.215	------	-420	259721
M75-12B	0.375	0.318	0.179	0.215	------	597738	259721
M75-12C	0.375	0.293	0.179	0.215	451648	------	259721
M75-13	0.375	0.327	0.179	0.215	------	258020	137764
M75-14	0.375	0.346	0.179	0.215	------	1041226	137764
M75-15A	0.688	0.383	0.179	0.215	8817944	------	52000
M75-15	0.375	0.288	0.179	0.215	------	66195	137764
M75-15B	0.375	0.318	0.179	0.215	------	903108	122147
M75-16A(L/E)	0.375	0.340	0.179	0.215	527441	------	122147
M75-16A(S/E)	0.375	0.318	0.151	0.198	401219	------	122147
M75-16	0.375	0.322	0.179	0.215	------	598838	259721
M75-17(U/S)	0.375	0.353	0.179	0.215	167217	------	259721
M75-17(D/S)	0.375	0.361	0.179	0.215	176938	------	259721
M75-17(BR.)	0.375	0.279	0.179	0.215	99000000	------	259721
M75-18A	0.375	0.308	0.179	0.215	657356	------	259721
M75-18B	0.375	0.293	0.179	0.215	451648	------	259721
M75-18C	0.375	0.338	0.179	0.215	------	910971	259721
M75-18D	0.375	0.293	0.179	0.215	451648	------	259721
M75-30(U/S)	0.375	0.427	0.179	0.215	599813	------	259721
M75-30(D/S)	0.375	0.448	0.179	0.215	750931	------	259721
M75-19-5604	0.375	0.055	0.191	0.215	-156148	------	259721
M75-19A	0.375	0.294	0.179	0.215	293733	------	259721
M75-20-5514	0.375	0.026	0.191	0.215	-164741	------	259721
M75-20A	0.375	0.341	0.179	0.215	29672118	------	26000
M75-21-5516	0.375	0.026	0.191	0.215	-164741	------	259721
M75-22	0.375	0.351	0.179	0.215	------	505190	128856
M75-23(U/S)	0.375	0.360	0.179	0.215	187612	------	259721
M75-23(D/S)	0.375	0.358	0.179	0.215	185018	------	259721
M75-23(BR.)	0.375	0.397	0.179	0.215	99000000	------	259721
M75-23A	0.375	0.321	0.179	0.215	752718	------	128856
M75-24	0.375	0.299	0.179	0.215	------	194894	128856
M75-25	0.375	0.294	0.179	0.215	175907	------	259721
M75-26	0.375	0.284	0.179	0.215	159189	------	137764
M75-27	0.375	0.317	0.179	0.215	346324	------	137764
M75-28	0.375	0.326	0.179	0.215	------	255985	137764
M75-29A	0.375	0.244	0.179	0.215	25866	------	259721
M75-29	0.375	0.288	0.179	0.215	66195	------	259721
M75-29B	0.375	0.285	0.179	0.215	72006	------	259721

===>Grouped by Line: 5TH PT ES CROSS "T" (2ES-1), No Sorting.

Note:
[1] Predictions are based on last Tmeas to analysis ending period.

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 14:02:33
Analysis Date: 02-DEC-2002 Time: 10:09:50
CHECWORKS FAC Version 1.0G (Build 75)

*** Wear Rate Analysis: Combined Rankings for Inspection ***

Run Name: Ext. Steam 5th Point
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.268

Duty Factor (Global): 1.000
Exclude Measure Wear: No

Component Name	Geometry Code	Average Wear Rate (mils/year)	Component Predicted Time to Tcrit (hrs) Non-Inspected	Inspected
M75-28	2	5.450	------	255985
M75-16A(S/E)	18	3.573	401219	------
M75-07-5517	25	11.769	-164741	------
M75-29A	52	17.797	25866	------
M75-02(U/S)	13	13.199	42051	------
M75-06-5515	25	11.769	-164741	------
M75-29	9	16.913	66195	------
M75-20-5514	25	11.769	-164741	------
M75-29B	9	15.053	72006	------
M75-15	9	14.822	------	66195
M75-21-5516	25	11.769	-164741	------
M75-17(D/S)	13	13.199	176938	------
M75-19-5604	22	10.789	-156148	------
M75-05-5603	22	10.789	-156148	------
M75-02(D/S)	13	13.199	84583	------
M75-12A	54	5.644	------	-420
M75-17(U/S)	13	13.199	167217	------
M75-23(D/S)	12	12.368	185018	------
M75-23(U/S)	12	12.368	187612	------
M75-26	2	5.450	159189	------
M75-25	54	7.242	175907	------
M75-09(U/S)	12	9.704	511441	------
M75-09(D/S)	12	9.704	327289	------
M75-30(U/S)	15	5.659	599813	------
M75-24	4	5.307	------	194894
M75-05A	58	4.316	------	223128
M75-11	4	5.450	------	290485
M75-13	2	5.450	------	258020
M75-19A	58	4.316	293733	------
M75-30(D/S)	15	4.981	750931	------
M75-27	52	3.700	346324	------
M75-08	58	4.316	------	367697
M75-10	62	1.765	------	430811
M75-12C	9	2.768	451648	------
M75-03C	9	2.768	451648	------
M75-22	58	3.297	------	505190
M75-18B	9	2.768	451648	------
M75-16	68	2.876	------	598838
M75-18D	9	2.768	451648	------
M75-01	68	2.876	------	572545
M75-03E	9	2.768	451648	------
M75-16A(L/E)	18	2.834	527441	------
M75-03D	9	2.768	------	777385
M75-03B	9	2.768	------	597738
M75-12B	9	2.768	------	597738
M75-18A	63	2.265	657356	------
M75-03A	63	2.265	------	951585
M75-23A	62	1.719	752718	------
M75-18C	9	2.157	------	910971
M75-15B	9	1.366	------	903108
M75-14	2	1.580	------	1041226
M75-15A	52	0.169	8817944	------
M75-04(U/S)	15	0.057	33872132	------
M75-20A	58	0.037	29672118	------
M75-01A(S/E)	18	0.056	37717044	------
M75-04(D/S)	15	0.050	41359316	------
M75-01A(L/E)	18	0.045	48918472	------
M75-06A	58	0.043	38075700	------
M75-17(BR.)	13	0.000	99000000	------
M75-23(BR.)	12	0.000	99000000	------
M75-09(BR.)	12	0.000	99000000	------
M75-02(BR.)	13	0.000	99000000	------

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 14:02:38
Analysis Date: 02-DEC-2002 Time: 10:09:50
CHECWORKS FAC Version 1.0G (Build 75)

****
*** Wear Rate Analysis: Wear Predictions Report ***
****

Run Name: Ext. Steam 5th Point
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.268

Duty Factor (Global): 1.000
Exclude Measure Wear: No

Component Name	Total Lifetime Wear (mils) Prd.[1]	Total Lifetime Wear (mils) Meas.	In-Service Cmp. Wear (mils) Prd.[1]	In-Service Cmp. Wear (mils) Meas.	In-Service Cmp. Tmeas (in)[3]	Method [2]	Time (hrs)[3]	In-Service Cmp. Thickness(mils)[4] Tp	In-Service Cmp. Thickness(mils)[4] Tm	Incremental Wear(mils)[5] PRWEAR	Time(hrs) Last Inspected
===>Grouped by Line: 5TH POINT ES (1ES-1), No Sorting.											
M75-01A(L/E)	73.0	41.0	0.0	0.0	0.398	MT	166985	375.0	398.0	0.4	0
M75-01A(S/E)	91.8	157.0	0.0	0.0	0.376	MT	166985	375.0	376.0	0.5	0
M75-01	66.0	67.0	66.0	67.0	0.337	MT	166985	309.0	337.0	19.3	166985
M75-03A	61.6	55.0	61.6	55.0	0.355	MT	220721	313.4	355.0	5.6	220721
M75-03B	75.2	78.0	75.2	78.0	0.325	MT	220721	299.8	325.0	6.9	220721
M75-03D	75.2	62.0	75.2	62.0	0.356	MT	220721	299.8	356.0	6.9	220721
M75-05A	99.0	78.0	99.0	78.0	0.304	MT	166985	276.0	304.0	29.0	166985
M75-06A	99.0	186.0	0.0	0.0	0.377	MT	166985	375.0	377.0	0.5	0
M75-08	105.9	94.0	105.9	94.0	0.336	MT	182553	269.1	336.0	22.1	182553
M75-10	22.2	94.0	22.2	94.0	0.281	MT	220721	352.8	281.0	5.6	220721
M75-11	68.5	58.0	68.5	58.0	0.358	MT	220721	306.5	358.0	17.2	220721
M75-12A	158.2	151.0	158.2	151.0	0.224	MT	233721	216.8	224.0	9.1	233721
M75-12B	75.2	119.0	75.2	119.0	0.325	MT	220721	299.8	325.0	6.9	220721
M75-13	74.4	69.0	74.4	69.0	0.338	MT	233721	300.6	338.0	11.3	233721
M75-14	21.6	41.0	21.6	41.0	0.349	MT	233721	353.4	349.0	3.3	233721
M75-15	174.7	172.0	174.7	172.0	0.346	MT	207721	200.3	346.0	58.4	207721
M75-15B	16.1	54.0	16.1	54.0	0.321	MT	233721	358.9	321.0	3.0	233721
M75-16A(L/E)	73.0	45.0	0.0	0.0	0.371	MT	152457	375.0	371.0	31.1	0
M75-16A(S/E)	91.8	104.0	0.0	0.0	0.357	MT	152457	375.0	357.0	39.2	0
M75-16	61.7	73.0	61.7	73.0	0.346	MT	152457	313.3	346.0	23.6	152457
M75-18C	60.5	58.0	60.5	58.0	0.341	MT	233721	314.5	341.0	3.5	233721
M75-22	105.9	164.0	26.5	51.0	0.373	MT	182553	348.5	373.0	22.1	182553
M75-24	159.4	176.0	31.5	98.0	0.346	MT	166985	343.5	346.0	46.6	166985
M75-28	62.6	50.0	62.6	50.0	0.349	MT	207721	312.4	349.0	23.1	207721
===>Grouped by Line: 5TH PT ES CROSS "T" (2ES-1), No Sorting.											

Notes:
[1] Predictions are for the time of last inspection (last known meas. wear).
[2] GW = Tmeas is minimum thickness from Band, Blanket or Area Method of greatest wear.
MT = Tmeas is component minimum thickness.
PW = Tmeas is Tinit - predicted wear.
US = Tmeas is user specified.
[3] If no Tmeas has been determined from measured data, then Tmeas = Tinit and Time = current component installation time.
Tmeas is used to determine Predicted Thickness and Component Predicted Time to Tcrit.
[4] These two values are used for thickness plot.
Tp = Predicted thickness at Tmeas.
Tm = Last measured thickness (Tmeas).
[5] PRWEAR = Incremental wear from last Tmeas time to analysis ending period.

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 14:02:45
Analysis Date: 02-DEC-2002 Time: 10:09:50
CHECWORKS FAC Version 1.0G (Build 75)

****

*** Wear Rate Analysis: Combined Summary Report ***

****

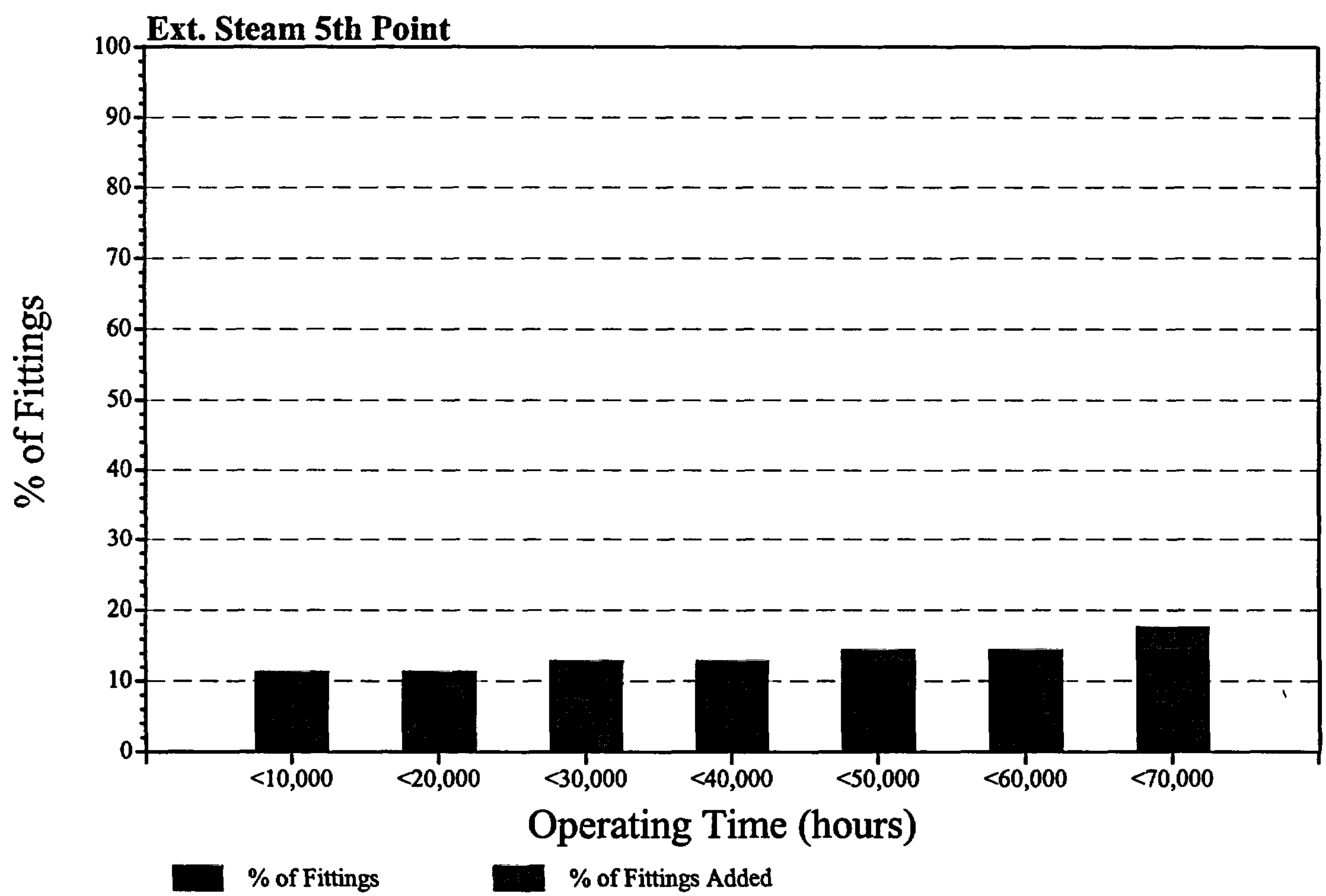
Run Name: Ext. Steam 5th Point
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.268

Duty Factor (Global): 1.000
Exclude Measure Wear: No

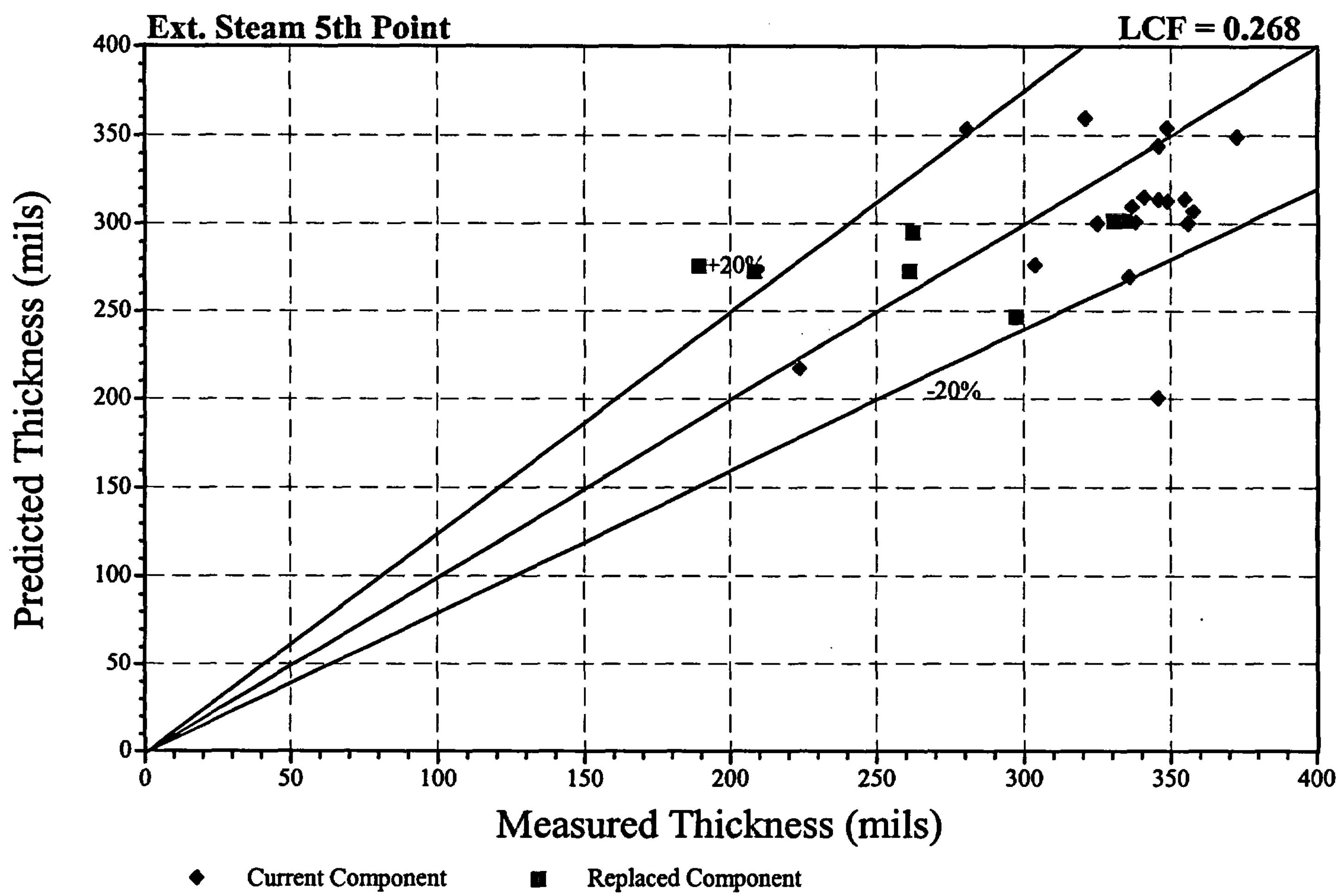
Component Name	Geom. Code	Average Wear Rate (mils/year)	Current Wear Rate (mils/year)	Thickness (in) Init.	Thickness (in) Prd.[1]	Thickness (in) Thoop	Thickness (in) Tcrit	Component Predict[1] Time to Tcrit (hrs) Non-Insp.	Component Predict[1] Time to Tcrit (hrs) Insp.	Total Lifetime Wear (mils) Prd.[2]	Total Lifetime Wear (mils) Meas.	In-Se Wea Prd.[
===>Grouped by Line: 5TH POINT ES (1ES-1), No Sorting.												
M75-01A(L/E)	18	0.045	0.033	0.375	0.398	0.179	0.215	48918472	------	73.0	41.0	---
M75-01A(S/E)	18	0.056	0.041	0.375	0.375	0.151	0.198	37717044	------	91.8	157.0	---
M75-01	68	2.876	1.571	0.375	0.318	0.179	0.215	------	572545	66.0	67.0	66.
M75-02(U/S)	13	13.199	7.209	0.375	0.250	0.179	0.215	42051	------	---	---	---
M75-02(D/S)	13	13.199	7.209	0.375	0.285	0.179	0.215	84583	------	---	---	---
M75-02(BR.)	13	0.000	0.000	0.375	0.281	0.179	0.215	99000000	------	---	---	---
M75-03A	63	2.265	1.237	0.375	0.349	0.179	0.215	------	951585	61.6	55.0	61.
M75-03B	9	2.768	1.512	0.375	0.318	0.179	0.215	------	597738	75.2	78.0	75.
M75-03C	9	2.768	1.512	0.375	0.293	0.179	0.215	451648	------	---	---	---
M75-03D	9	2.768	1.512	0.375	0.349	0.179	0.215	------	777385	75.2	62.0	75.
M75-03E	9	2.768	1.512	0.375	0.293	0.179	0.215	451648	------	---	---	---
M75-04(U/S)	15	0.057	0.049	0.375	0.403	0.179	0.215	33872132	------	---	---	---
M75-04(D/S)	15	0.050	0.043	0.375	0.417	0.179	0.215	41359316	------	---	---	---
M75-05-5603	22	10.789	5.893	0.375	0.055	0.191	0.215	-156148	------	---	---	---
M75-05A	58	4.316	2.357	0.375	0.275	0.179	0.215	------	223128	99.0	78.0	99.
M75-06-5515	25	11.769	6.428	0.375	0.026	0.191	0.215	-164741	------	---	---	---
M75-06A	58	0.043	0.037	0.375	0.377	0.179	0.215	38075700	------	99.0	186.0	---
M75-07-5517	25	11.769	6.428	0.375	0.026	0.191	0.215	-164741	------	---	---	---
M75-08	58	4.316	2.357	0.375	0.314	0.179	0.215	------	367697	105.9	94.0	105.
M75-09(U/S)	12	9.704	6.755	0.375	0.609	0.179	0.215	511441	------	---	---	---
M75-09(D/S)	12	9.704	6.755	0.375	0.467	0.179	0.215	327289	------	---	---	---
M75-09(BR.)	12	0.000	0.000	0.375	0.371	0.179	0.215	99000000	------	---	---	---
M75-10	62	1.765	1.229	0.375	0.275	0.179	0.215	------	430811	22.2	94.0	22.
M75-11	4	5.450	3.794	0.375	0.341	0.179	0.215	------	290485	68.5	58.0	68.
M75-12A	54	5.644	3.082	0.375	0.215	0.179	0.215	------	-420	158.2	151.0	158.
M75-12B	9	2.768	1.512	0.375	0.318	0.179	0.215	------	597738	75.2	119.0	75.
M75-12C	9	2.768	1.512	0.375	0.293	0.179	0.215	451648	------	---	---	---
M75-13	2	5.450	3.794	0.375	0.327	0.179	0.215	------	258020	74.4	69.0	74.
M75-14	2	1.580	1.100	0.375	0.346	0.179	0.215	------	1041226	21.6	41.0	21.
M75-15A	52	0.169	0.167	0.688	0.383	0.179	0.215	8817944	------	---	---	---
M75-15	9	14.822	9.612	0.375	0.288	0.179	0.215	------	66195	174.7	172.0	174.
M75-15B	9	1.366	0.999	0.375	0.318	0.179	0.215	------	903108	16.1	54.0	16.
M75-16A(L/E)	18	2.834	2.074	0.375	0.340	0.179	0.215	527441	------	73.0	45.0	---
M75-16A(S/E)	18	3.573	2.615	0.375	0.318	0.151	0.198	401219	------	91.8	104.0	---
M75-16	68	2.876	1.571	0.375	0.322	0.179	0.215	------	598838	61.7	73.0	61.
M75-17(U/S)	13	13.199	7.209	0.375	0.353	0.179	0.215	167217	------	---	---	---
M75-17(D/S)	13	13.199	7.209	0.375	0.361	0.179	0.215	176938	------	---	---	---
M75-17(BR.)	13	0.000	0.000	0.375	0.279	0.179	0.215	99000000	------	---	---	---
M75-18A	63	2.265	1.237	0.375	0.308	0.179	0.215	657356	------	---	---	---
M75-18B	9	2.768	1.512	0.375	0.293	0.179	0.215	451648	------	---	---	---
M75-18C	9	2.157	1.178	0.375	0.338	0.179	0.215	------	910971	60.5	58.0	60.
M75-18D	9	2.768	1.512	0.375	0.293	0.179	0.215	451648	------	---	---	---
M75-30(U/S)	15	5.659	3.090	0.375	0.427	0.179	0.215	599813	------	---	---	---
M75-30(D/S)	15	4.981	2.720	0.375	0.448	0.179	0.215	750931	------	---	---	---
M75-19-5604	22	10.789	5.893	0.375	0.055	0.191	0.215	-156148	------	---	---	---
M75-19A	58	4.316	2.357	0.375	0.294	0.179	0.215	293733	------	---	---	---
M75-20-5514	25	11.769	6.428	0.375	0.026	0.191	0.215	-164741	------	---	---	---
M75-20A	58	0.037	0.037	0.375	0.341	0.179	0.215	29672118	------	---	---	---
M75-21-5516	25	11.769	6.428	0.375	0.026	0.191	0.215	-164741	------	---	---	---
M75-22	58	3.297	2.357	0.375	0.351	0.179	0.215	------	505190	105.9	164.0	26.
M75-23(U/S)	12	12.368	6.755	0.375	0.360	0.179	0.215	187612	------	---	---	---
M75-23(D/S)	12	12.368	6.755	0.375	0.358	0.179	0.215	185018	------	---	---	---
M75-23(BR.)	12	0.000	0.000	0.375	0.397	0.179	0.215	99000000	------	---	---	---
M75-23A	62	1.719	1.229	0.375	0.321	0.179	0.215	752718	------	---	---	---
M75-24	4	5.307	3.794	0.375	0.299	0.179	0.215	------	194894	159.4	176.0	31.
M75-25	54	7.242	3.955	0.375	0.294	0.179	0.215	175907	------	---	---	---
M75-26	2	5.450	3.794	0.375	0.284	0.179	0.215	159189	------	---	---	---
M75-27	52	3.700	2.575	0.375	0.317	0.179	0.215	346324	------	---	---	---
M75-28	2	5.450	3.794	0.375	0.326	0.179	0.215	------	255985	62.6	50.0	62.
M75-29A	52	17.797	9.939	0.375	0.244	0.179	0.215	25866	------	---	---	---
M75-29	9	16.913	9.612	0.375	0.288	0.179	0.215	66195	------	---	---	---
M75-29B	9	15.053	8.536	0.375	0.285	0.179	0.215	72006	------	---	---	---
===>Grouped by Line: 5TH PT ES CROSS "T" (2ES-1), No Sorting.												

Notes:
[1] Predictions are based on last Tmeas to analysis ending period.
[2] Predictions are for the time of last inspection (last known meas. wear).
[3] GW = Tmeas is minimum thickness from Band, Blanket or Area Method of greatest wear.
MT = Tmeas is component minimum thickness.
PW = Tmeas is Tinit - predicted wear.
US = Tmeas is user specified.
[4] If no Tmeas has been determined from measured data, then Tmeas = Tinit and Time = current component installation time.
Tmeas is used to determine Predicted Thickness and Component Predicted Time to Tcrit.

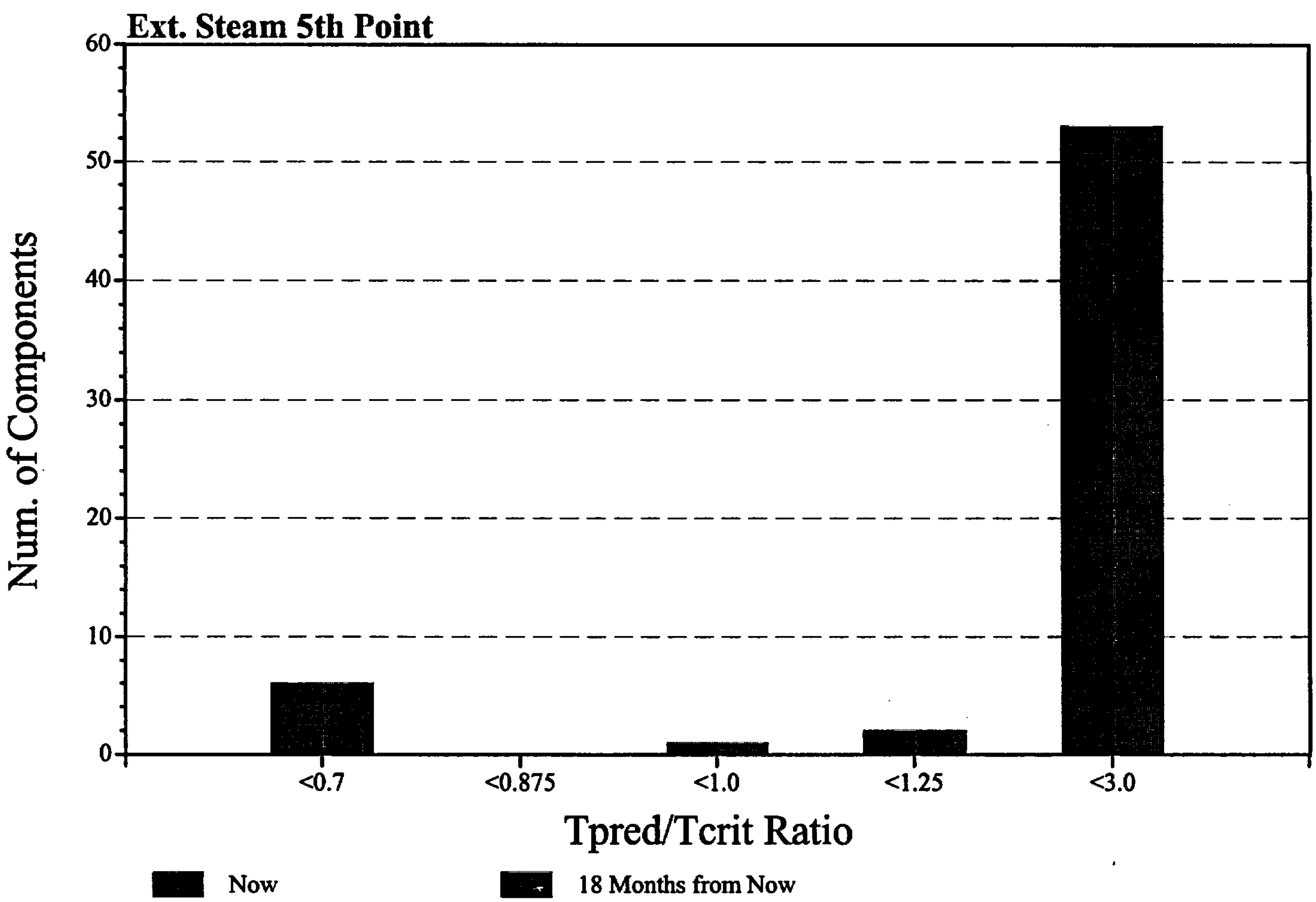
Cumulative % of Comp. Time to Tcrit

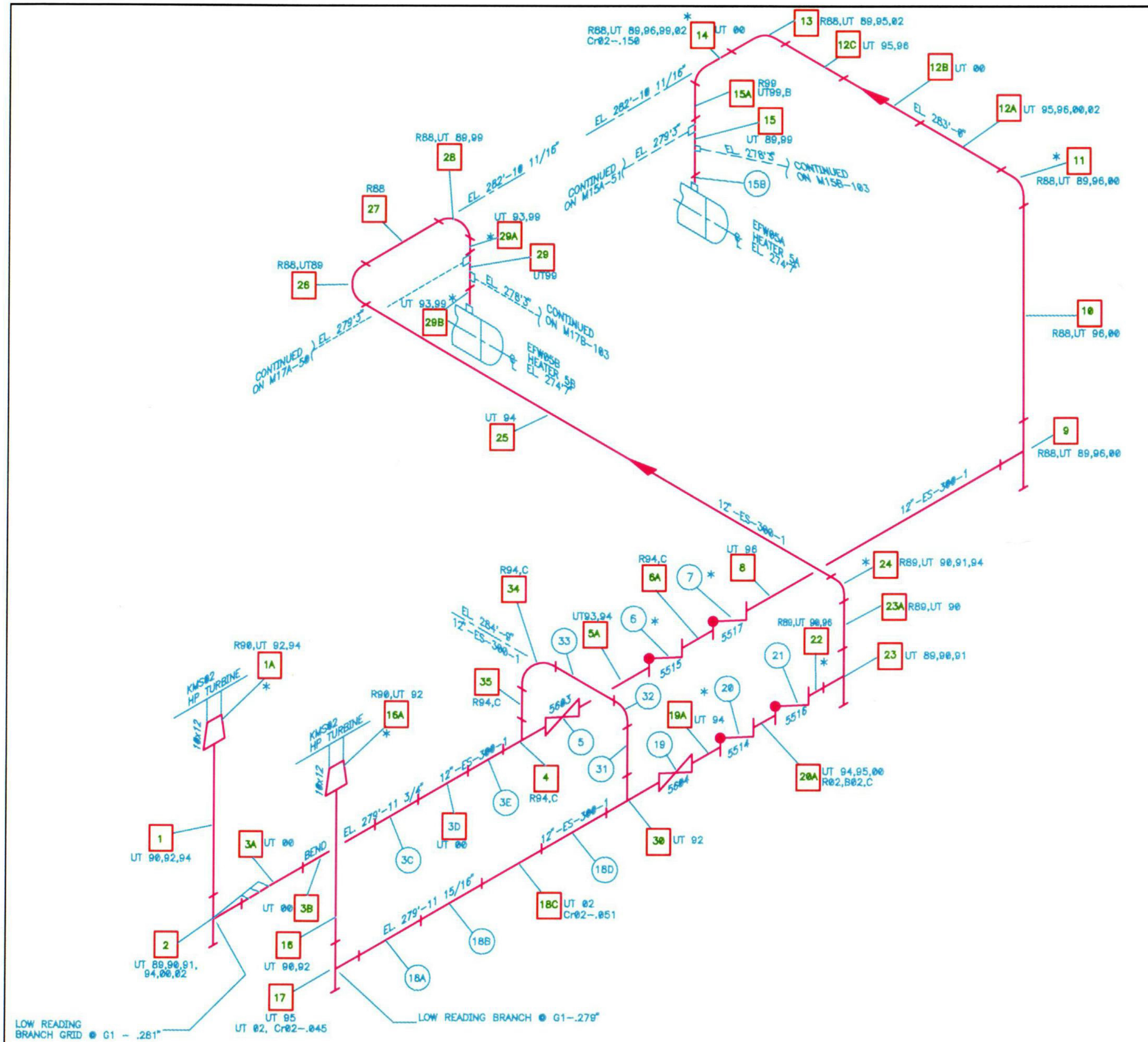


Comparison of Thickness Predictions



Tpred/Tcrit Ratio Plot





REF. DWG.S

PIPING D–304–041
PIPING D–304–042
P&ID 33013–1903
WALKDOWN SKETCH (RG&E LETTER 13N1–RO–L0324 DATED 5/4/90)

NORTH

○ – DENOTES EROSION CORROSION COMPONENT I.D.

□ – DENOTES COMPONENT WITH CHECWORKS HISTORY

* – DENOTES FITTING WHICH HAS ADJACENT FITTING LOCATED WITHIN 1 DIAMETER UPSTREAM

UT – ULTRASONIC TESTING

RT – RADIOGRAPHY TESTING

R – DENOTES RFO WHEN COMPONENT WAS REPLACED

C – DENOTES COMPONENT AS CHROME–MOLY

B – DENOTES BASELINE UT INSPECTION

– DRAWING TO BE USED FOR COMPONENT IDENTIFICATION PURPOSES ONLY

CHECWORKS LINE/RUN NAME:
5th POINT ES (1ES–1)/EXT STEAM 5th POINT
5th POINT ES CROSS "T" (2ES–1)/EXT STEAM 5th POINT

△					
△					
3	MARKING CHECWORKS COMPONENT HISTORY	AJV3 8/30/02			
2	AS–BUILT	AJV3 7/8/97	AB 7/8/97	AB 7/8/97	MJS 7/10/98
NUMBER	REVISION	DRAWN BY	CHECKED	RESP. ENG.	REVIEWER

	CONSTRUCTION	
	LIMITED CONSTRUCTION: AS NOTED	
	PRELIMINARY NOT FOR CONSTRUCTION	
	BIDDING PURPOSES	
	INSPECTION	
DATE	RELEASED FOR	ENGR.

Ginna Station (FACILITY) | 1M0521 (CAD FILENAME) | ROCHESTER GAS & ELECTRIC CORP. ROCHESTER, NEW YORK

SCALE NONE

EROSION CORROSION ISI PROGRAM

ISOMETRIC

STEAM EXTRACTION TO 5A & 5B HEATERS

JOB NO.	FIGURE	DRAWING NO.	REV.
EWR 4499	M–75	10904–521	3

C02

Company : Rochester Gas and Electric
Plant : R. E. Ginna
Unit : 0
DB Name : GINNA

Report Date : 27-MAY-03 Time : 13:51:40

CHECWORKS FAC Version 1.0G (Build 75)

****
*** UT Summary ***
****

Component : M21-39A
Line : PRE-SEP TANK HTR 4 (1ES-5)
Geometry Type : 45-DEG ELBOW

===> Section : U/S Main
Tnom = 0.375 (in), Tinit = 0.375 (in), Tscreen = 0.172 (in)

Period(s)	Grid Size (RxC)	No. of Points	Avg. Thk.	Standard Dev.	Min. Thk. (RxC)	Max. Thk. (RxC)	Total Life Wear	Wear Method	Total Svr. Hours.
RFO 1990	9x17	150	0.377	0.020	0.338(3,Q)	0.429(3,I)	0.048	Blanket	15617.0
RFO 1999	9x17	153	0.338	0.052	0.232(3,Q)	0.427(3,I)	0.155	Blanket	85764.0
RFO 2000	9x17	153	0.335	0.056	0.226(3,Q)	0.474(1,L)	0.174	Blanket	98764.0
RFO 2002	9x17	153	0.331	0.058	0.212(3,Q)	0.427(3,I)	0.177	Blanket	111764.0

===> Section : U/S Ext.
Tnom = 0.375 (in), Tinit = 0.000 (in), Tscreen = 0.172 (in)

Period(s)	Grid Size (RxC)	No. of Points	Avg. Thk.	Standard Dev.	Min. Thk. (RxC)	Max. Thk. (RxC)	Total Life Wear	Wear Method	Total Svr. Hours.
RFO 2000	3x17	51	0.334	0.017	0.299(3,I)	0.363(1,N)	0.060	Band	----
RFO 2002	3x17	51	0.331	0.019	0.294(3,I)	0.366(1,N)	0.081	Band	----

===> Section : D/S Ext.
Tnom = 0.375 (in), Tinit = 0.000 (in), Tscreen = 0.172 (in)

Period(s)	Grid Size (RxC)	No. of Points	Avg. Thk.	Standard Dev.	Min. Thk. (RxC)	Max. Thk. (RxC)	Total Life Wear	Wear Method	Total Svr. Hours.
RFO 2000	7x17	119	0.340	0.039	0.247(1,A)	0.392(4,G)	0.130	Band	----
RFO 2002	7x17	119	0.337	0.041	0.235(1,A)	0.396(3,H)	0.141	Band	----

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 13:54:17
Analysis Date: 02-DEC-2002 Time: 14:31:13
CHECWORKS FAC Version 1.0G (Build 75)

****
*** Wear Rate Analysis: Wear Rates/Input Data Report ***
****

Run Name: PreSep Tank to Htr 4
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.899

Duty Factor (Global): 1.000
Exclude Measure Wear: No

Component Name	Geom. Code	Average Wear Rate (mils/year)	Current Wear Rate (mils/year)	Temp. (F)	Velocity (ft/s)	Steam Quality	Diameter (in)
===>Grouped by Line: PRE-SEP TANK HTR 4 (1ES-5), No Sorting.							
M21-23	31	0.210	0.207	352.3	35.081	0.950	16.000
M21-24	4	0.138	0.127	352.3	35.081	0.950	16.000
M21-25	54	0.144	0.133	352.3	35.081	0.950	16.000
M21-26	4	0.138	0.127	352.3	35.081	0.950	16.000
M21-27	54	0.144	0.133	352.3	35.081	0.950	16.000
M21-28	2	0.138	0.127	352.3	35.081	0.950	16.000
M21-29	52	0.093	0.086	352.3	35.081	0.950	16.000
M21-30(U/S)	15	9.719	6.580	352.3	35.081	0.950	16.000
M21-30(D/S)	15	8.554	5.791	352.3	35.081	0.950	16.000
M21-31-5602	22	17.769	12.030	352.3	59.546	0.950	16.000
M21-32	58	0.082	0.076	352.3	59.546	0.950	16.000
M21-32A	58	0.066	0.061	352.3	35.081	0.950	16.000
M21-33(U/S)	12	0.000	0.000	0.0	0.000	0.000	16.000
M21-33(D/S)	12	0.244	0.227	352.3	35.081	0.950	16.000
M21-33(BR.)	12	0.162	0.150	352.3	35.081	0.950	16.000
M21-34	62	0.045	0.041	352.3	35.275	0.950	16.000
M21-35	4	0.137	0.127	352.3	35.081	0.950	16.000
M21-36A	54	9.076	8.422	352.3	35.081	0.950	16.000
M21-36B	54	11.447	8.422	352.3	35.081	0.950	16.000
M21-36C	9	4.217	3.102	352.3	35.081	0.950	16.000
M21-36D	9	4.217	3.102	352.3	35.081	0.950	16.000
M21-37	2	0.129	0.127	352.3	35.081	0.950	16.000
M21-37A	52	7.453	5.483	352.3	35.081	0.950	16.000
M21-38	1	6.981	6.917	352.3	35.081	0.950	16.000
M21-39	51	6.559	4.825	352.3	35.081	0.950	16.000
M21-39A	1	8.844	6.917	352.3	35.081	0.950	16.000
M21-39B	51	6.170	4.825	352.3	35.081	0.950	16.000
M21-40	2	10.328	8.077	352.3	35.081	0.950	16.000
M21-41	52	5.871	4.592	352.3	44.781	0.950	16.000
===>Grouped by Line: PRE-SEP TK TO HTR 4 (1ES-3), No Sorting.							
M22-28	31	0.228	0.210	352.3	36.584	0.950	16.000
M22-29	61	0.161	0.149	352.3	35.275	0.950	16.000
M22-29A	61	0.220	0.149	352.3	35.275	0.950	16.000
M22-30	2	0.138	0.127	352.3	35.081	0.950	16.000
M22-31	52	0.093	0.086	352.3	35.081	0.950	16.000
M22-32	2	0.138	0.127	352.3	35.081	0.950	16.000
M22-32A	52	0.093	0.086	352.3	35.081	0.950	16.000
M22-32B	52	0.093	0.086	352.3	35.081	0.950	16.000
M22-33(U/S)	12	0.000	0.000	0.0	0.000	0.000	16.000
M22-33(D/S)	12	0.248	0.230	352.3	36.584	0.950	16.000
M22-33(BR.)	12	0.162	0.150	352.3	35.081	0.950	16.000
M22-34	62	0.045	0.041	352.3	35.275	0.950	16.000
M22-35	4	0.137	0.127	352.3	35.081	0.950	16.000
M22-36	54	9.076	8.422	352.3	35.081	0.950	16.000
M22-37(U/S)	15	9.719	6.580	352.3	35.081	0.950	16.000
M22-37(D/S)	15	8.554	5.791	352.3	35.081	0.950	16.000
M22-37A	65	11.559	6.491	352.3	35.081	0.950	16.000
M22-37B	9	5.524	3.102	352.3	35.081	0.950	16.000
M22-38	2	9.835	8.077	352.3	35.081	0.950	16.000
M22-39	52	0.086	0.086	352.3	35.081	0.950	16.000
M22-40	2	0.129	0.127	352.3	35.081	0.950	16.000
M22-41-5601	22	15.805	10.702	352.3	69.970	0.950	16.000
M22-42	58	7.622	4.281	352.3	69.970	0.950	16.000

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 13:55:07
Analysis Date: 02-DEC-2002 Time: 14:31:13
CHECWORKS FAC Version 1.0G (Build 75)

*** Wear Rate Analysis: Inspection History Report ***

Run Name: PreSep Tank to Htr 4
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.899

Duty Factor (Global): 1.000
Exclude Measure Wear: No

===>Grouped by Line: PRE-SEP TANK HTR 4 (1ES-5), No Sorting.

Component Name	Geom. Code	Material No.	Material Cr. (%)	Material Cu. (%)	Material Mo. (%)	Sigma (psi)	Time (hrs) Last Inspected	Time (hrs) Replaced	Analysis Option	Measured Wear (mils)
M21-23	31	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	31	21	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	31	21	0.00	0.00	0.00	15000	159681	159681		174
*Replacement #3	31	18	1.90	0.00	0.87	15000	-----	207721		---
M21-24	4	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	4	21	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	4	21	0.00	0.00	0.00	15000	152457	159681		152
M21-25	54	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	54	5	0.00	0.00	0.00	15000	137574	137574		283
*Replacement #2	54	26	1.90	0.00	0.87	15000	-----	159681		---
M21-26	4	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	4	21	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	4	21	0.00	0.00	0.00	15000	152457	159681		98
M21-27	54	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	54	5	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	54	5	0.00	0.00	0.00	15000	152457	159681		180
M21-28	2	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	2	21	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	2	21	0.00	0.00	0.00	15000	152457	159681		143
M21-29	52	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	52	5	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	52	5	0.00	0.00	0.00	15000	152457	159681		141
M21-30(U/S)	15	21	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	15	21	0.00	0.00	0.00	15000	-----	86864		---
M21-30(D/S)	15	21	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	15	21	0.00	0.00	0.00	15000	-----	86864		---
M21-31-5602	22	93	0.00	0.00	0.00	14000	-----	-----		---
*Replacement #1	22	93	0.00	0.00	0.00	14000	-----	86864		---
M21-32	58	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	58	5	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	58	5	0.00	0.00	0.00	15000	-----	166985		---
M21-32A	58	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	58	5	0.00	0.00	0.00	15000	-----	166985		---
M21-33(U/S)	12	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	12	21	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	12	21	0.00	0.00	0.00	15000	-----	166985		---
M21-33(D/S)	12	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	12	21	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	12	21	0.00	0.00	0.00	15000	-----	166985		---
M21-33(BR.)	12	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	12	21	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	12	21	0.00	0.00	0.00	15000	-----	166985		---
M21-34	62	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	62	5	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	62	5	0.00	0.00	0.00	15000	-----	166985		---
M21-35	4	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	4	21	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	4	21	0.00	0.00	0.00	15000	159681	166985		152
M21-36A	54	5	0.03	0.00	0.00	15000	233721	-----		85
*Replacement #1	54	5	0.00	0.00	0.00	15000	-----	166985	Excl LCF	---
M21-36B	54	5	0.00	0.00	0.00	15000	207721	-----		180
M21-36C	9	5	0.00	0.00	0.00	15000	-----	-----		---
M21-36D	9	5	0.00	0.00	0.00	15000	220721	-----		70
*Replacement #1	54	5	0.00	0.00	0.00	15000	-----	106373	Excl LCF	---
M21-37	2	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	2	18	1.90	0.00	0.87	15000	-----	207721		---
M21-37A	52	5	0.00	0.00	0.00	15000	137574	-----		34
*Replacement #1	52	5	0.00	0.00	0.00	15000	-----	106373	Excl LCF	---
M21-38	1	21	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	1	21	0.00	0.00	0.00	15000	-----	106373		---
*Replacement #2	1	21	0.00	0.00	0.00	15000	207721	220721		173
M21-39	51	5	0.00	0.00	0.00	15000	137574	-----		40
*Replacement #1	51	5	0.00	0.00	0.00	15000	-----	106373	Excl LCF	---
M21-39A	1	21	0.00	0.00	0.00	15000	233721	-----		177
*Replacement #1	1	21	0.00	0.00	0.00	15000	-----	121957	Excl LCF	---
M21-39B	51	5	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	51	5	0.00	0.00	0.00	15000	-----	121957		---
M21-40	2	21	0.00	0.00	0.00	15000	207721	-----		104
*Replacement #1	3	21	0.00	0.00	0.00	15000	-----	121957	Excl LCF	---
M21-41	52	5	0.00	0.00	0.00	15000	-----	-----		---
*Replacement #1	53	5	0.00	0.00	0.00	15000	-----	121957		---

===>Grouped by Line: PRE-SEP TK TO HTR 4 (1ES-3), No Sorting.

Component Name	Geom. Code	Material No.	Material Cr. (%)	Material Cu. (%)	Material Mo. (%)	Sigma (psi)	Time (hrs) Last Inspected	Time (hrs) Replaced	Analysis Option	Measured Wear (mils)
M22-28	31	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	31	21	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	31	21	0.00	0.00	0.00	15000	152457	159681		203
M22-29	61	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	61	5	0.00	0.00	0.00	15000	137574	137574		227
*Replacement #2	61	5	0.00	0.00	0.00	15000	152457	159681		55
M22-29A	61	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	61	5	0.00	0.00	0.00	15000	-----	86864		---

M22-30	2	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	2	21	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	2	21	0.00	0.00	0.00	15000	152457	159681		140
M22-31	52	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	52	5	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	52	5	0.00	0.00	0.00	15000	-----	159681		---
M22-32	2	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	2	21	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	2	21	0.00	0.00	0.00	15000	159681	159681		115
M22-32A	52	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	52	5	0.00	0.00	0.00	15000	-----	86864		---
*Replacement #2	52	5	0.00	0.00	0.00	15000	159681	166985		133
M22-32B	52	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	52	5	0.00	0.00	0.00	15000	-----	166985		---
M22-33(U/S)	12	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	12	21	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	12	21	0.00	0.00	0.00	15000	-----	166985		---
M22-33(D/S)	12	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	12	21	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	12	21	0.00	0.00	0.00	15000	-----	166985		---
M22-33(BR.)	12	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	12	21	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	12	21	0.00	0.00	0.00	15000	-----	166985		---
M22-34	62	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	62	5	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	62	5	0.00	0.00	0.00	15000	-----	166985		---
M22-35	4	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	4	21	0.00	0.00	0.00	15000	-----	121957		---
*Replacement #2	4	21	0.00	0.00	0.00	15000	-----	166985		---
M22-36	54	5	0.00	0.00	0.00	15000	233721	-----		37
*Replacement #1	54	5	0.00	0.00	0.00	15000	-----	166985	Excl LCF	---
M22-37(U/S)	15	21	0.00	0.00	0.00	15000	144871	-----		92
*Replacement #1	15	21	0.00	0.00	0.00	15000	-----	86864	Excl LCF	---
M22-37(D/S)	15	21	0.00	0.00	0.00	15000	144871	-----		85
*Replacement #1	15	21	0.00	0.00	0.00	15000	-----	86864	Excl LCF	---
M22-37A	65	5	0.03	0.00	0.00	15000	233721	-----		203
M22-37B	9	5	0.00	0.00	0.00	15000	137574	-----		91
M22-38	2	21	0.00	0.00	0.00	15000	207721	-----		59
*Replacement #1	2	21	0.00	0.00	0.00	15000	137574	137574		197
M22-39	52	26	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	52	5	0.00	0.00	0.00	15000	220721	233721		120
M22-40	2	18	1.90	0.00	0.87	15000	-----	-----		---
*Replacement #1	2	18	1.90	0.00	0.87	15000	-----	207721		---
M22-41-5601	22	93	0.00	0.00	0.00	14000	-----	-----		---
*Replacement #1	22	93	0.00	0.00	0.00	14000	-----	86864		---
M22-42	58	5	0.00	0.00	0.00	15000	207721	-----		127

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 13:55:18
Analysis Date: 02-DEC-2002 Time: 14:31:13
CHECWORKS FAC Version 1.0G (Build 75)

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*** Wear Rate Analysis: Thickness/Service Time Report ***

****

Run Name: PreSep Tank to Htr 4
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.899

Duty Factor (Global): 1.000
Exclude Measure Wear: No

Component Name	Thickness (in) Init.	Prd.[1]	Thoop	Tcrit	Component Predicted[1] Time to Tcrit (hrs) Non-Inspected	Inspected	Component Actual Service Time (hrs)
===>Grouped by Line: PRE-SEP TANK HTR 4 (1ES-5), No Sorting.							
M21-23	0.375	0.374	0.093	0.172	8518705	------	52000
M21-24	0.375	0.379	0.093	0.172	14268658	------	100040
M21-25	0.375	0.347	0.093	0.172	11569330	------	100040
M21-26	0.375	0.351	0.093	0.172	12342572	------	100040
M21-27	0.375	0.360	0.093	0.172	12426999	------	100040
M21-28	0.375	0.366	0.093	0.172	13374404	------	100040
M21-29	0.375	0.376	0.093	0.172	20664146	------	100040
M21-30(U/S)	0.375	0.417	0.093	0.172	326544	------	172857
M21-30(D/S)	0.375	0.451	0.093	0.172	421570	------	172857
M21-31-5602	0.375	0.024	0.100	0.172	-99426	------	172857
M21-32	0.375	0.356	0.093	0.172	21260100	------	92736
M21-32A	0.375	0.374	0.093	0.172	29114606	------	92736
M21-33(U/S)	0.375	0.588	0.093	0.172	99000000	------	92736
M21-33(D/S)	0.375	0.637	0.093	0.172	17975818	------	92736
M21-33(BR.)	0.375	0.521	0.093	0.093	24938120	------	92736
M21-34	0.375	0.333	0.093	0.172	34013292	------	92736
M21-35	0.375	0.404	0.093	0.172	15927829	------	92736
M21-36A	0.375	0.286	0.093	0.172	------	118581	92736
M21-36B	0.375	0.171	0.093	0.172	------	-94068	153348
M21-36C	0.375	0.301	0.093	0.172	364766	------	153348
M21-36D	0.375	0.319	0.093	0.172	------	415249	153348
M21-37	0.375	0.416	0.093	0.172	16800514	------	52000
M21-37A	0.375	0.249	0.093	0.172	------	122857	153348
M21-38	0.375	0.406	0.093	0.172	296248	------	39000
M21-39	0.375	0.272	0.093	0.172	------	181676	153348
M21-39A	0.375	0.191	0.093	0.172	------	24658	137764
M21-39B	0.375	0.251	0.093	0.172	143553	------	137764
M21-40	0.375	0.255	0.093	0.172	------	90433	137764
M21-41	0.375	0.283	0.093	0.172	211127	------	137764
===>Grouped by Line: PRE-SEP TK TO HTR 4 (1ES-3), No Sorting.							
M22-28	0.500	0.497	0.093	0.213	11838205	------	100040
M22-29	0.375	0.367	0.093	0.172	11489262	------	100040
M22-29A	0.375	0.371	0.093	0.172	11698876	------	172857
M22-30	0.375	0.382	0.093	0.172	14475023	------	100040
M22-31	0.375	0.352	0.093	0.172	18232262	------	100040
M22-32	0.375	0.443	0.093	0.172	18610588	------	100040
M22-32A	0.375	0.337	0.093	0.172	16720578	------	92736
M22-32B	0.375	0.374	0.093	0.172	20469730	------	92736
M22-33(U/S)	0.500	0.644	0.093	0.213	99000000	------	92736
M22-33(D/S)	0.500	0.497	0.093	0.093	15400687	------	92736
M22-33(BR.)	0.375	0.373	0.093	0.093	16322375	------	92736
M22-34	0.375	0.329	0.093	0.172	33165758	------	92736
M22-35	0.375	0.435	0.093	0.172	18060278	------	92736
M22-36	0.375	0.331	0.093	0.172	------	165388	92736
M22-37(U/S)	0.375	0.420	0.093	0.172	------	330442	172857
M22-37(D/S)	0.375	0.421	0.093	0.093	------	496597	172857
M22-37A	0.375	0.153	0.093	0.172	------	-26000	259721
M22-37B	0.375	0.247	0.093	0.172	------	212705	259721
M22-38	0.375	0.281	0.093	0.172	------	118631	122147
M22-39	0.375	0.349	0.093	0.172	17909134	------	26000
M22-40	0.375	0.439	0.093	0.172	18382654	------	52000
M22-41-5601	0.375	0.063	0.100	0.172	-84972	------	172857
M22-42	0.375	0.222	0.093	0.172	------	102796	259721

Note:
[1] Predictions are based on last Tmeas to analysis ending period.

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 13:55:27
Analysis Date: 02-DEC-2002 Time: 14:31:13
CHECWORKS FAC Version 1.0G (Build 75)

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*** Wear Rate Analysis: Combined Rankings for Inspection ***

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Run Name: PreSep Tank to Htr 4
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.899

Duty Factor (Global): 1.000
Exclude Measure Wear: No

Component Name	Geometry Code	Average Wear Rate (mils/year)	Component Predicted Time to Tcrit (hrs) Non-Inspected	Inspected
M21-39A	1	8.844	------	24658
M21-40	2	10.328	------	90433
M21-31-5602	22	17.769	-99426	------
M22-37A	65	11.559	------	-26000
M21-30(U/S)	15	9.719	326544	------
M22-41-5601	22	15.805	-84972	------
M21-36B	54	11.447	------	-94068
M22-38	2	9.835	------	118631
M22-37(U/S)	15	9.719	------	330442
M22-42	58	7.622	------	102796
M21-36A	54	9.076	------	118581
M22-36	54	9.076	------	165388
M21-37A	52	7.453	------	122857
M21-39B	51	6.170	143553	------
M22-37(D/S)	15	8.554	------	496597
M21-30(D/S)	15	8.554	421570	------
M21-39	51	6.559	------	181676
M21-41	52	5.871	211127	------
M22-37B	9	5.524	------	212705
M21-38	1	6.981	296248	------
M21-36C	9	4.217	364766	------
M21-36D	9	4.217	------	415249
M22-33(D/S)	12	0.248	15400687	------
M21-23	31	0.210	8518705	------
M21-33(D/S)	12	0.244	17975818	------
M22-29	61	0.161	11489262	------
M22-28	31	0.228	11838205	------
M21-25	54	0.144	11569330	------
M22-29A	61	0.220	11698876	------
M21-33(BR.)	12	0.162	24938120	------
M21-26	4	0.138	12342572	------
M22-33(BR.)	12	0.162	16322375	------
M21-27	54	0.144	12426999	------
M21-28	2	0.138	13374404	------
M21-24	4	0.138	14268658	------
M22-30	2	0.138	14475023	------
M21-35	4	0.137	15927829	------
M22-32	2	0.138	18610588	------
M22-32A	52	0.093	16720578	------
M21-37	2	0.129	16800514	------
M22-35	4	0.137	18060278	------
M22-39	52	0.086	17909134	------
M22-40	2	0.129	18382654	------
M22-31	52	0.093	18232262	------
M21-29	52	0.093	20664146	------
M22-32B	52	0.093	20469730	------
M21-32	58	0.082	21260100	------
M21-32A	58	0.066	29114606	------
M21-34	62	0.045	34013292	------
M22-34	62	0.045	33165758	------
M22-33(U/S)	12	0.000	99000000	------
M21-33(U/S)	12	0.000	99000000	------

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 13:55:33
Analysis Date: 02-DEC-2002 Time: 14:31:13
CHECWORKS FAC Version 1.0G (Build 75)

*** Wear Rate Analysis: Wear Predictions Report ***

Run Name: PreSep Tank to Htr 4
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.899

Duty Factor (Global): 1.000
Exclude Measure Wear: No

Component Name	Total Lifetime Wear (mils) Prd.[1]	Meas.	In-Service Cmp. Wear (mils) Prd.[1]	Meas.	In-Service Cmp. Tmeas,Method,Time (in)[3]	[2]	(hrs)[3]	In-Service Cmp. Thickness(mils)[4] Tp	Tm	Incremental Wear(mils)[5] PRWEAR	Time(hrs) Last Inspected
===>Grouped by Line: PRE-SEP TANK HTR 4 (1ES-5), No Sorting.											
M21-23	225.2	174.0	0.0	0.0	0.375	--	207721	375.0	375.0	1.2	0
M21-24	128.2	152.0	0.0	0.0	0.381	MT	159681	375.0	381.0	1.6	0
M21-25	301.6	283.0	0.0	0.0	0.349	MT	159681	375.0	349.0	1.6	0
M21-26	128.2	98.0	0.0	0.0	0.353	MT	159681	375.0	353.0	1.6	0
M21-27	133.6	180.0	0.0	0.0	0.362	MT	159681	375.0	362.0	1.6	0
M21-28	128.2	143.0	0.0	0.0	0.368	MT	159681	375.0	368.0	1.6	0
M21-29	87.0	141.0	0.0	0.0	0.377	MT	159681	375.0	377.0	1.1	0
M21-35	62.7	152.0	0.0	0.0	0.405	MT	166985	375.0	405.0	1.5	0
M21-36A	71.1	85.0	71.1	85.0	0.311	MT	233721	303.9	311.0	25.0	233721
M21-36B	149.7	180.0	149.7	180.0	0.222	MT	207721	225.3	222.0	50.7	207721
M21-36D	59.9	70.0	59.9	70.0	0.333	MT	220721	315.1	333.0	13.9	220721
M21-37A	37.4	34.0	37.4	34.0	0.342	MT	137574	337.6	342.0	93.1	137574
M21-38	123.0	173.0	0.0	0.0	0.437	MT	220721	375.0	437.0	31.1	0
M21-39	32.9	41.0	32.9	41.0	0.354	MT	137574	342.1	354.0	81.9	137574
M21-39A	118.6	177.0	118.6	177.0	0.212	MT	233721	256.4	212.0	20.5	233721
M21-40	113.8	104.0	113.8	104.0	0.304	MT	207721	261.2	304.0	48.6	207721
===>Grouped by Line: PRE-SEP TK TO HTR 4 (1ES-3), No Sorting.											
M22-28	211.8	203.0	0.0	0.0	0.500	--	159681	500.0	500.0	2.6	0
M22-29	372.8	282.0	0.0	0.0	0.368	MT	207721	375.0	368.0	0.9	0
M22-30	128.2	140.0	0.0	0.0	0.384	MT	159681	375.0	384.0	1.6	0
M22-32	135.7	115.0	0.0	0.0	0.444	MT	166985	375.0	444.0	1.5	0
M22-32A	92.1	133.0	0.0	0.0	0.338	MT	166985	375.0	338.0	1.0	0
M22-36	71.1	37.0	71.1	37.0	0.356	MT	233721	303.9	356.0	25.0	233721
M22-37(U/S)	92.0	92.0	92.0	92.0	0.520	MT	144871	283.0	520.0	99.8	144871
M22-37(D/S)	81.0	85.0	81.0	85.0	0.509	MT	144871	294.0	509.0	87.8	144871
M22-37A	323.4	203.0	323.4	203.0	0.172	MT	233721	51.6	172.0	19.3	233721
M22-37B	111.1	91.0	111.1	91.0	0.300	MT	137574	263.9	300.0	52.7	137574
M22-38	377.8	256.0	88.5	59.0	0.330	MT	207721	286.5	330.0	48.6	207721
M22-39	264.9	120.0	0.0	0.0	0.349	MT	233721	375.0	349.0	0.3	0
M22-42	200.2	127.0	200.2	127.0	0.248	MT	207721	174.8	248.0	25.8	207721

Notes:
[1] Predictions are for the time of last inspection (last known meas. wear).
[2] GW = Tmeas is minimum thickness from Band, Blanket or Area Method of greatest wear.
MT = Tmeas is component minimum thickness.
PW = Tmeas is Tinit - predicted wear.
US = Tmeas is user specified.
[3] If no Tmeas has been determined from measured data, then Tmeas = Tinit and Time = current component installation tim
Tmeas is used to determine Predicted Thickness and Component Predicted Time to Tcrit.
[4] These two values are used for thickness plot.
Tp = Predicted thickness at Tmeas.
Tm = Last measured thickness (Tmeas).
[5] PRWEAR = Incremental wear from last Tmeas time to analysis ending period.

Company: Rochester Gas and Electric
Plant: R. E. Ginna
Unit:
DB Name: GINNA

Report Date: 27-MAY-2003 Time: 13:55:39
Analysis Date: 02-DEC-2002 Time: 14:31:13
CHECWORKS FAC Version 1.0G (Build 75)

****
*** Wear Rate Analysis: Combined Summary Report ***
****

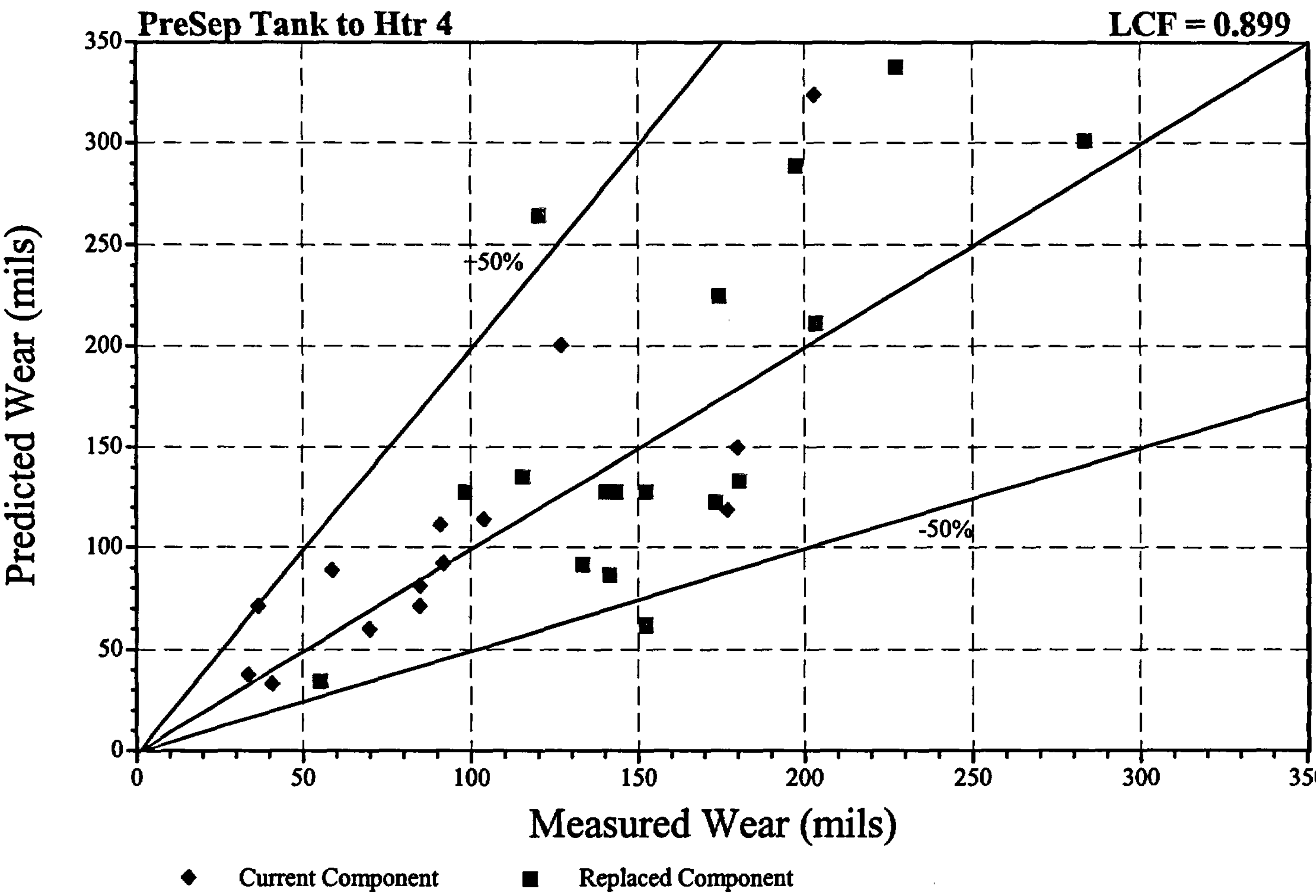
Run Name: PreSep Tank to Htr 4
Ending Period: RFO 2005
Total Plant Operating Hours: 259721
WRA Data Option: COMP->NFA
Line Correction Factor: 0.899

Duty Factor (Global): 1.000
Exclude Measure Wear: No

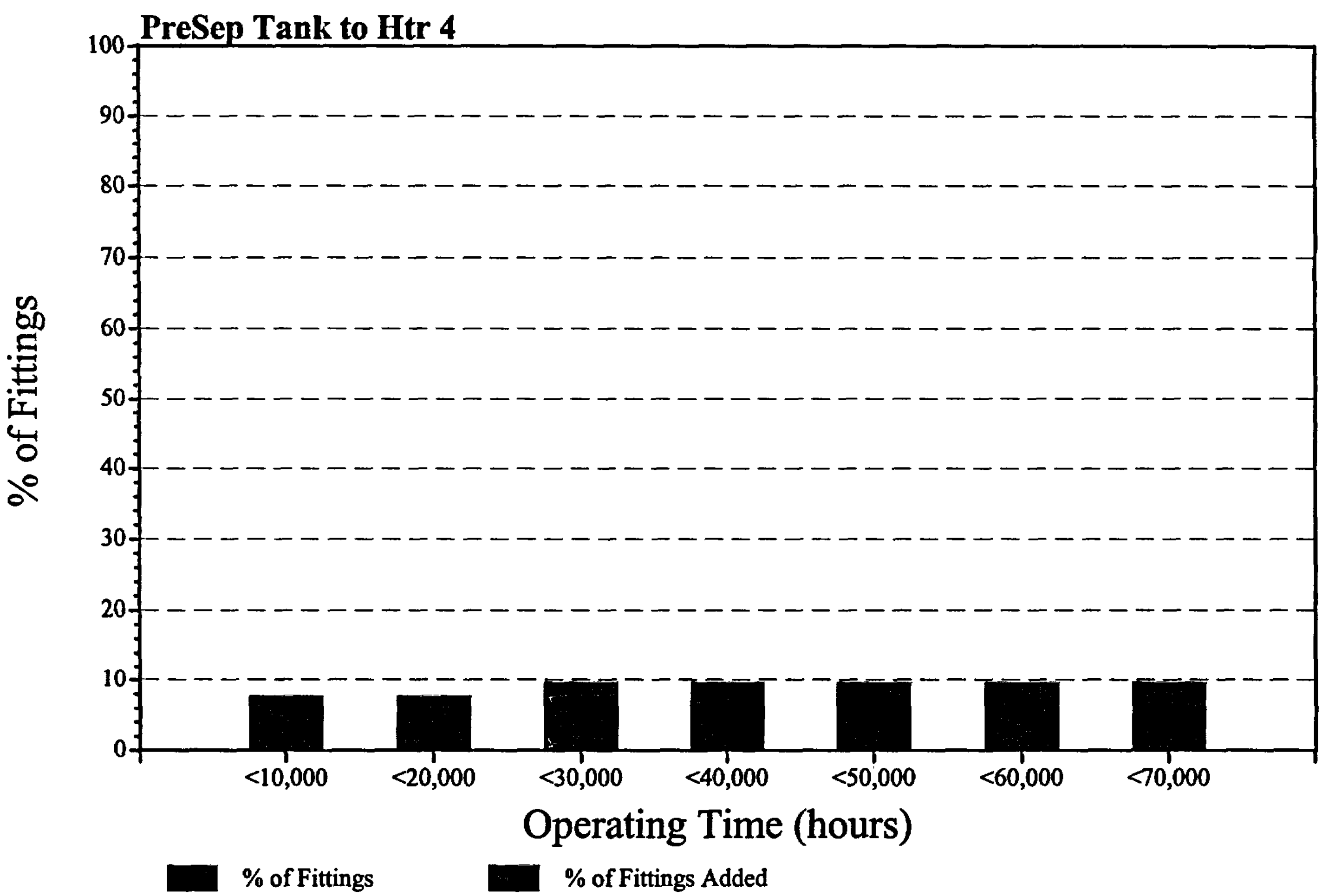
Component Name	Geom. Code	Average Wear Rate (mils/year)	Current Wear Rate (mils/year)	Thickness (in) Init.	Thickness (in) Prd.[1]	Thickness (in) Thoop	Thickness (in) Tcrit	Component Predict[1] Time to Tcrit (hrs) Non-Insp.	Component Predict[1] Time to Tcrit (hrs) Insp.	Total Lifetim Wear (mils) Prd.[2]	Total Lifetim Wear (mils) Mea
===>Grouped by Line: PRE-SEP TANK HTR 4 (1ES-5), No Sorting.											
M21-23	31	0.210	0.207	0.375	0.374	0.093	0.172	8518705	------	225.2	174
M21-24	4	0.138	0.127	0.375	0.379	0.093	0.172	14268658	------	128.2	152
M21-25	54	0.144	0.133	0.375	0.347	0.093	0.172	11569330	------	301.6	283
M21-26	4	0.138	0.127	0.375	0.351	0.093	0.172	12342572	------	128.2	98
M21-27	54	0.144	0.133	0.375	0.360	0.093	0.172	12426999	------	133.6	180
M21-28	2	0.138	0.127	0.375	0.366	0.093	0.172	13374404	------	128.2	143
M21-29	52	0.093	0.086	0.375	0.376	0.093	0.172	20664146	------	87.0	141
M21-30(U/S)	15	9.719	6.580	0.375	0.417	0.093	0.172	326544	------	---	-
M21-30(D/S)	15	8.554	5.791	0.375	0.451	0.093	0.172	421570	------	---	-
M21-31-5602	22	17.769	12.030	0.375	0.024	0.100	0.172	-99426	------	---	-
M21-32	58	0.082	0.076	0.375	0.356	0.093	0.172	21260100	------	---	-
M21-32A	58	0.066	0.061	0.375	0.374	0.093	0.172	29114606	------	---	-
M21-33(U/S)	12	0.000	0.000	0.375	0.588	0.093	0.172	99000000	------	---	-
M21-33(D/S)	12	0.244	0.227	0.375	0.637	0.093	0.172	17975818	------	---	-
M21-33(BR.)	12	0.162	0.150	0.375	0.521	0.093	0.093	24938120	------	---	-
M21-34	62	0.045	0.041	0.375	0.333	0.093	0.172	34013292	------	---	-
M21-35	4	0.137	0.127	0.375	0.404	0.093	0.172	15927829	------	62.7	152
M21-36A	54	9.076	8.422	0.375	0.286	0.093	0.172	------	118581	71.1	85
M21-36B	54	11.447	8.422	0.375	0.171	0.093	0.172	------	-94068	149.7	180
M21-36C	9	4.217	3.102	0.375	0.301	0.093	0.172	364766	------	---	-
M21-36D	9	4.217	3.102	0.375	0.319	0.093	0.172	------	415249	59.9	70
M21-37	2	0.129	0.127	0.375	0.416	0.093	0.172	16800514	------	---	-
M21-37A	52	7.453	5.483	0.375	0.249	0.093	0.172	------	122857	37.4	34
M21-38	1	6.981	6.917	0.375	0.406	0.093	0.172	296248	------	123.0	173
M21-39	51	6.559	4.825	0.375	0.272	0.093	0.172	------	181676	32.9	41
M21-39A	1	8.844	6.917	0.375	0.191	0.093	0.172	------	24658	118.6	177
M21-39B	51	6.170	4.825	0.375	0.251	0.093	0.172	143553	------	---	-
M21-40	2	10.328	8.077	0.375	0.255	0.093	0.172	------	90433	113.8	104
M21-41	52	5.871	4.592	0.375	0.283	0.093	0.172	211127	------	---	-
===>Grouped by Line: PRE-SEP TK TO HTR 4 (1ES-3), No Sorting.											
M22-28	31	0.228	0.210	0.500	0.497	0.093	0.213	11838205	------	211.8	203
M22-29	61	0.161	0.149	0.375	0.367	0.093	0.172	11489262	------	372.8	282
M22-29A	61	0.220	0.149	0.375	0.371	0.093	0.172	11698876	------	---	-
M22-30	2	0.138	0.127	0.375	0.382	0.093	0.172	14475023	------	128.2	140
M22-31	52	0.093	0.086	0.375	0.352	0.093	0.172	18232262	------	---	-
M22-32	2	0.138	0.127	0.375	0.443	0.093	0.172	18610588	------	135.7	115
M22-32A	52	0.093	0.086	0.375	0.337	0.093	0.172	16720578	------	92.1	133
M22-32B	52	0.093	0.086	0.375	0.374	0.093	0.172	20469730	------	---	-
M22-33(U/S)	12	0.000	0.000	0.500	0.644	0.093	0.213	99000000	------	---	-
M22-33(D/S)	12	0.248	0.230	0.500	0.497	0.093	0.093	15400687	------	---	-
M22-33(BR.)	12	0.162	0.150	0.375	0.373	0.093	0.093	16322375	------	---	-
M22-34	62	0.045	0.041	0.375	0.329	0.093	0.172	33165758	------	---	-
M22-35	4	0.137	0.127	0.375	0.435	0.093	0.172	18060278	------	---	-
M22-36	54	9.076	8.422	0.375	0.331	0.093	0.172	------	165388	71.1	37
M22-37(U/S)	15	9.719	6.580	0.375	0.420	0.093	0.172	------	330442	92.0	92
M22-37(D/S)	15	8.554	5.791	0.375	0.421	0.093	0.093	------	496597	81.0	85
M22-37A	65	11.559	6.491	0.375	0.153	0.093	0.172	------	-26000	323.4	203
M22-37B	9	5.524	3.102	0.375	0.247	0.093	0.172	------	212705	111.1	91
M22-38	2	9.835	8.077	0.375	0.281	0.093	0.172	------	118631	377.8	256
M22-39	52	0.086	0.086	0.375	0.349	0.093	0.172	17909134	------	264.9	120
M22-40	2	0.129	0.127	0.375	0.439	0.093	0.172	18382654	------	---	-
M22-41-5601	22	15.805	10.702	0.375	0.063	0.100	0.172	-84972	------	---	-
M22-42	58	7.622	4.281	0.375	0.222	0.093	0.172	------	102796	200.2	127

Notes:
[1] Predictions are based on last Tmeas to analysis ending period.
[2] Predictions are for the time of last inspection (last known meas. wear).
[3] GW = Tmeas is minimum thickness from Band, Blanket or Area Method of greatest wear.
MT = Tmeas is component minimum thickness.
PW = Tmeas is Tinit - predicted wear.
US = Tmeas is user specified.
[4] If no Tmeas has been determined from measured data, then Tmeas = Tinit and Time = current component installation tim
Tmeas is used to determine Predicted Thickness and Component Predicted Time to Tcrit.

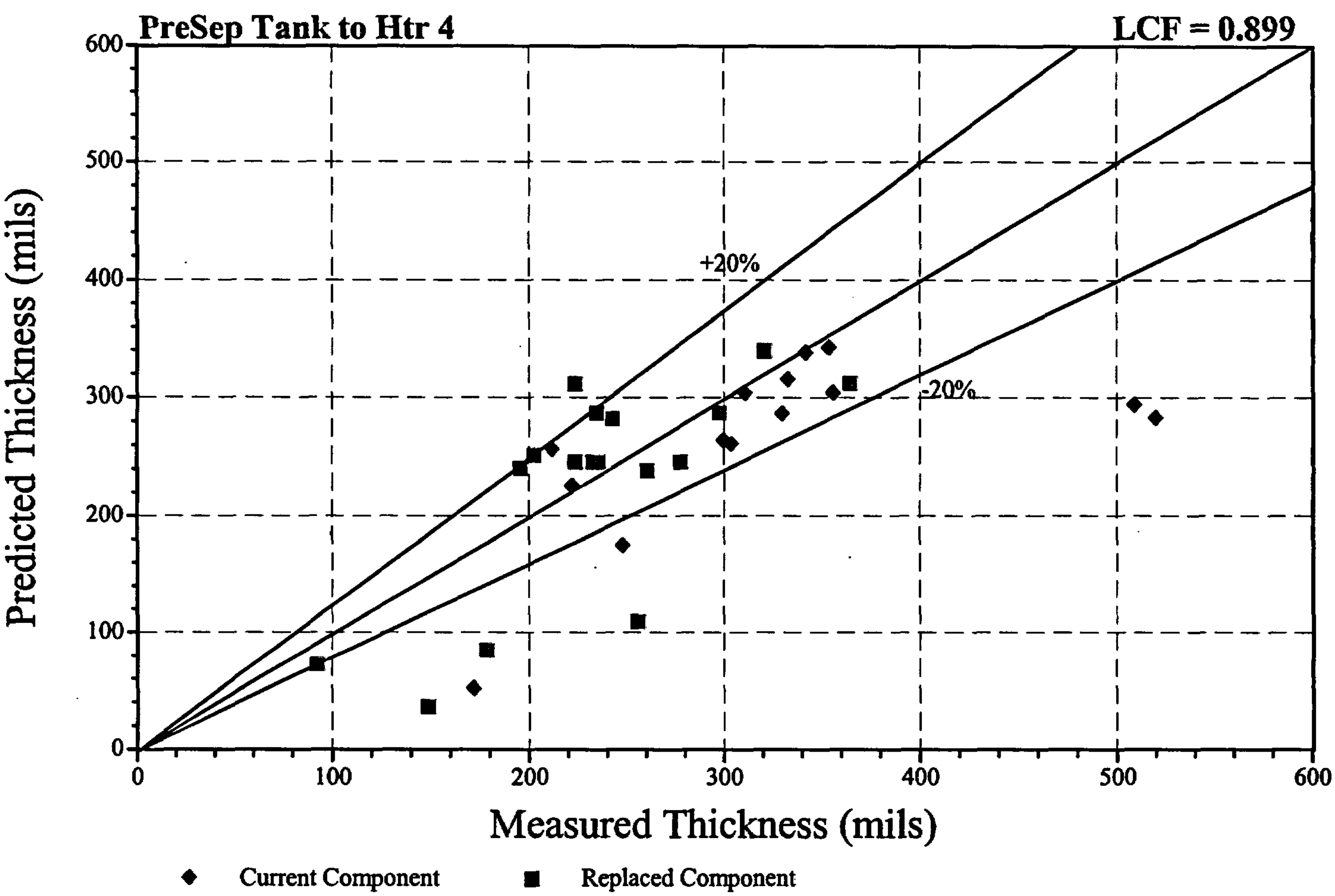
Comparison of Wear Predictions



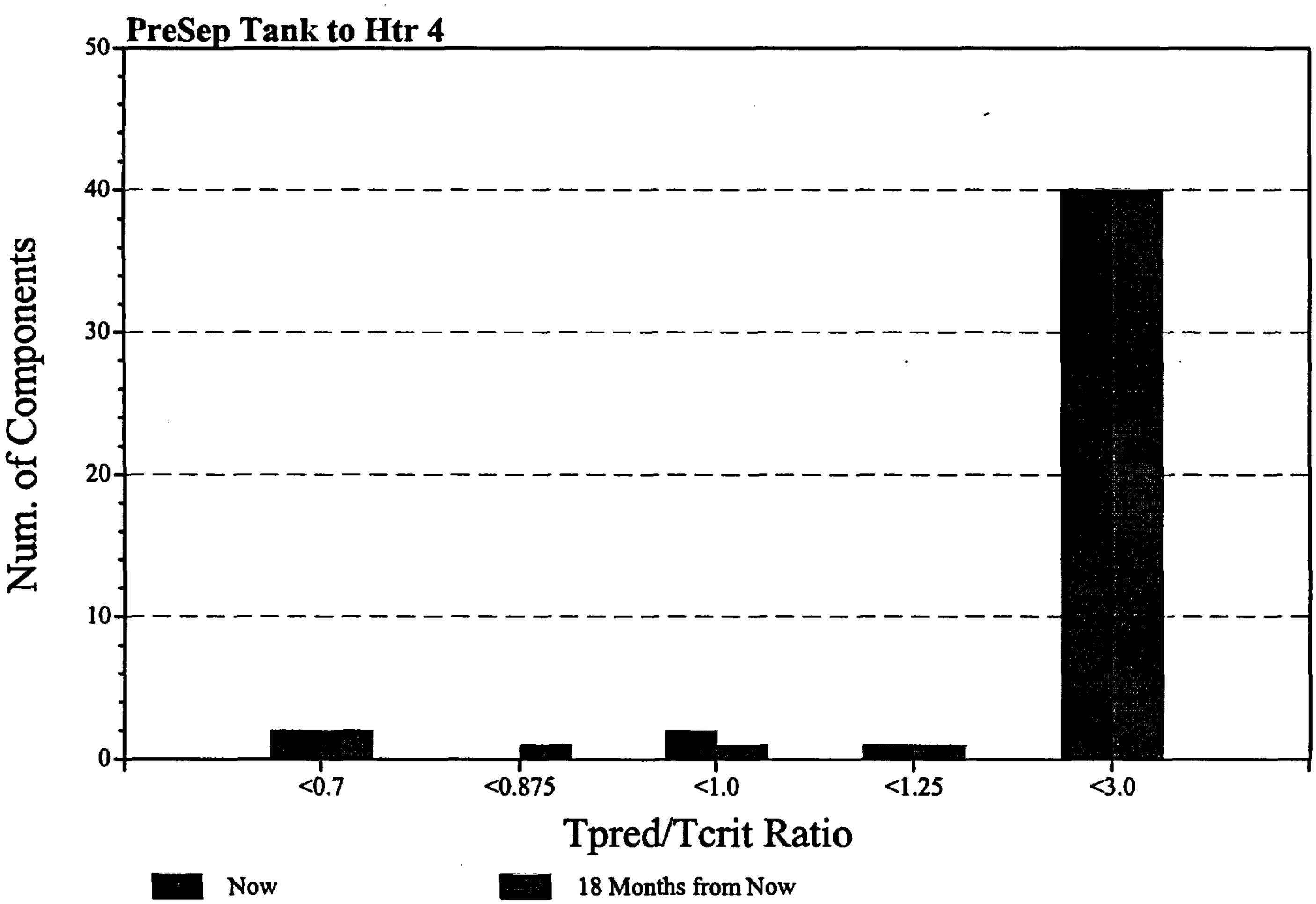
Cumulative % of Comp. Time to Tcrit



Comparison of Thickness Predictions



Tpred/Tcrit Ratio Plot



ATTACHMENT 4
CLARIFICATION FOR RAIs 4.2.1-1 AND 4.2.2-1 INCLUDED IN RG&E DESIGN ANALYSIS DA-ME-2003-024

EVALUATION OF REACTOR VESSEL BELTLINE WELDS' RT_{NDT} FOR PRESSURIZED THERMAL SHOCK DURING PERIOD OF EXTENDED OPERATION

Ginna Station

Rochester Gas and Electric Corporation
89 East Avenue
Rochester, New York 14649

DA - ME - 2003 - 024

Revision (0)

Prepared by: ______________________ 6/9/03
Assigned Engineer — Date

Reviewed by: ______________________ 6/10/03
Independent Reviewer — Date

REVISION STATUS SHEET

Revision Number	Affected Sections	Description of Revision
0	All	Original issue

TABLE OF CONTENTS

Evaluation of Reactor Vessel Beltline Welds' RT_{NDT} for Pressurized Thermal Shock During Period of Extended Operation

1.0Purpose

This analysis evaluates the reference temperature of nil ductility transition (RT_{NDT}) of Ginna's reactor vessel beltline welds during the period of extended operation, to make sure that requirements of 10CFR50.61 are still satisfied. The beltline welds are the limiting materials for the vessel due to their chemical contents of Cu and Ni and location relative to the fuel core where these receive the largest fluence. The welds that are evaluated are SA-1101 and SA-847.

2.0 Conclusions

Considering the fluences that are predicted during the period of extended operation, the circumferential welds, SA-1101 and SA-847 still satisfy requirements of 10CFR50.61 for pressurized thermal shock. The adjusted reference temperature (ART), of both welds are still less than 300^0 F, which is the screening criterion for pressurized thermal shock in 10CFR50.61 for circumferential welds.

3.0Design Inputs

1. Projected fluence during period of extended operation at the weld locations are taken from WCAP - 15885 (Reference 4.1).
2. Chemistry factor of weld SA-847, which was calculated from available surveillance capsule data is also available from WCAP - 15885 (Reference 4.1).
3. Conservative parameters that were used in calculating ART for SA-1101 were taken from References 4.2 (Reg. Guide 1.99, Rev. 2) and 4.3 (10CFR50.61).

4.0Referenced Documents

1. WCAP - 15885, Rev. 0, " R. E. Ginna Heat-up and Cool-down Limit Curves for Normal Operation ", July 2002
2. Regulatory Guide 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Materials", May 1988.
3. Ginna UFSAR, Section 5.3.1.2
4. BAW - 2425, Rev. 1, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessel of R. E. Ginna For Extended Life Through 54 Effective Full Power Years", June 2002.
5. Ginna UFSAR, Figure 5.3-2.

6. Telephone conference, RG&E and NRC (Barry Elliot et al.) on April 23, 2003.
7. 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Event"

5.0 Assumptions

Assumptions are made and justified in appropriate sections where these are utilized.

6.0 Computer Codes

None

7.0 Analysis

Reference 5 shows the locations of the circumferential welds SA-1101 and SA-847 with respect to the core. SA-1101 connects the nozzle shell to the intermediate shell and is the weld closest to the RV nozzle. It is located 10" above the top of the fuel core. SA-847 is located 14.8" below the centerline of the fuel core and is the limiting weld for the beltline materials as delineated in Reference 3 based on radiation exposure and chemical composition.

During a recent telephone conference with the NRC (Reference 6), a need to check the reference temperature of nil ductility transition (RT_{NDT}) of SA-1101 is required. This is due to the fact that the expected fluence at its location during the period of extended operation is greater than 10^{18} n/cm^2 (E>1MeV) as predicted in Reference 1. This value should be checked against the criterion for pressurized thermal shock given in Reference 7.

The RT_{NDT} for SA-847 will also be calculated and checked against the requirements of Reference 7, since it is the limiting weld for the Ginna RV. It is exposed to the highest fluence among the beltline welds per Reference 3.

7.1 Evaluation of RT_{NDT} for SA-1101

As a conservative approach, an envelope value of RT_{NDT} for SA-1101 will be calculated based on an assumption that the surveillance data and material composition for this weld are not available. Parameters suggested by Reg. Guide 1.99 (Reference 2) and 10CFR50.61 (Reference 7) will be utilized. Reference 3 (Table 5.3-4) identifies this weld as utilizing a Linde 80 Flux. This information will be utilized later to identify the initial RT_{NDT} and Margin parameters needed for evaluation the adjusted reference temperature, which accounts for the shift of RT_{NDT} due to radiation effects.

7.1.1 Use of Regulatory Guide 1.99 Procedures

For cases where surveillance data are not available, the use of Regulatory Positions 1.1 and 1.2 are subject to the limitation provisions that are delineated in Regulatory Position 1.3. These limitations are satisfied for SA-1101 as listed in the following findings:

- The RV shell forgings, which are SA-508, Class 2 per Reference 3 has a minimum yield strength which is greater than 50 ksi.
- The irradiation temperature is between 525 F and 590 F per Reference 4.
- The copper and nickel contents are within the ranges in Figure 1 and Tables 1 & 2 of Reg. Guide 1.99.

7.1.2 Calculation of the Adjusted Reference Temperature

Per Regulatory Position 1.1 of Reg. Guide 1.99, the Adjusted Reference Temperature is calculated using the expression,

$$ART = \text{Initial } RT_{NDT} + \Delta RT_{NDT} + \text{Margin} \qquad (1)$$

Where:

Initial RT_{NDT} = Referenced temperature for the unirradiated material as defined in Paragraph NB-2331 of Section III of the ASME Boiler and Pressure Vessel Code. If measured values for the material are not available, a generic mean value for that class of material may be used if there are sufficient test results to establish a mean and standard deviation for the class. SA-1101 belongs to the Linde 80 Flux class, which has generic mean value for the Initial RT_{NDT} of 0° F per Reference 7. Hence for SA-1101,

$$\text{Initial } RT_{NDT} = 0^{\circ}\text{F} \qquad (2)$$

ΔRT_{NDT} = Mean value of the adjustment in reference temperature caused by irradiation, which is calculated as follows,

$$\Delta RT_{NDT} = (CF)\, f^{(0.28 - 0.10 \log f)} \qquad (3)$$

CF = Chemistry factor given in Table 1 (RG 1.99) for welds.

f = neutron fluence at the inside surface of the RV, 10^{19} n/cm^2. Per Reference 1, this value at the inner surface of the RV where SA-1101 is located, at 54 EFPY is 0.198. However, this value will be doubled as agreed upon during the telephone conference with the NRC (Reference 6), to account for uncertainties.

$$f = 0.198 \times 2 = .396 \qquad (4)$$

Per Reg. Guide 1.99 (Reference 2), since surveillance data are assumed to be not available for SA-1101, we will also assume a composition of Cu and Ni as 0.35% and 1.00% respectively. From Table 1 of Reference 2, the chemistry factor for SA-1101 is,

$$CF = 272^\circ \text{ F} \qquad (5)$$

Substituting values in Equations 4 and 5 into Equation 3, we have,

$$\Delta RT_{NDT} = 272 \times (0.396)^{(0.28 - 0.10 \times \log 0.396)}$$

$$= 272 \times 0.74331 = 202.18^\circ \text{ F} \qquad (6)$$

Margin = A quantity that is added to obtain conservative, upper bound values of adjusted reference temperature for the calculations required by Appendix G to 10CFR50. Since a measured value of initial RT_{NDT} for SA-1101 is not available, a generic mean value for that class of material can be utilized. From Reference 7, this value is 66° F for welds. Hence,

$$\text{Margin} = 66^\circ \text{ F} \qquad (7)$$

Substituting values in Equations 2, 6, and 7 into Equation 1, gives

$$ART = 0 + 202.18 + 66 = 268.18^\circ \text{ F} \qquad (8)$$

This is the adjusted reference temperature of SA-1101 at the end of the extended period of operation (54 EFPY).

7.1.3 Comparison with PTS Screening Criterion in 10CFR50.61

For circumferential beltline welds, the screening criterion in 10CFR50.61 (Reference 7) against pressurized thermal shock events is 300° F. Since the adjusted reference temperature for SA-1101 at 54 EFPY is less than the criterion, i.e.,

$$268.18^\circ \text{ F} < 300^\circ \text{ F} \qquad (9)$$

This weld is NOT a concern for pressurized thermal events during the period of extended operation.

7.2 Evaluation of RT_{NDT} for SA-847

SA-847, being the limiting weld for the Ginna RV has surveillance data that are available. This data comes from four surveillance capsules that have already been pulled out and tested. Results of the tests are given in Reference 1. Using procedures described in Regulatory Position 2.1 of Reg. Guide 1.99 (Reference 2), the chemistry factor for SA-847 is,

$$CF_{SA\text{-}847} = 161.9^{\circ}\,F \qquad (10)$$

From Reference 1, the predicted fluence at the inner surface of the Ginna RV for 54 EFPY, was calculated to be,

$$f = 5.01 \quad (10^{19}\ n/cm^2, E>1\ Mev) \qquad (11)$$

Other parameters that are needed to calculate the adjusted reference temperature for SA-847 are given in Reference 1 as,

$$\text{Initial } RT_{NDT} = -4.8^{\circ}\,F \qquad (12)$$

$$\text{Margin} = 48.3^{\circ}\,F \qquad (13)$$

7.2.1 Calculate RT_{NDT} Using Regulatory Position 2.1

Per Position 2.1 of Reference 2, the adjusted reference temperature can be calculated using Equations 1 and 3.

Substituting values of the parameters of SA-847 from Equations 10, 11, 12 and 13 into Equations 1 and 3 gives,

$$ART = -4.8 + (161.9) \times 5.01^{(0.28 - 0.10 \times \log 5.01)} + 48.3$$

$$= -4.8 + 161.9 \times 1.4027 + 48.3$$

$$= -4.8 + 227.1 + 48.3$$

$$= 270.6^{\circ}\,F \qquad (14)$$

This is the adjusted reference temperature for SA-847 at 54 EFPY using Regulatory Position 2.1.

7.2.2 Calculate RT_{NDT} Using Regulatory Position 1.1

When surveillance data are available for belt-line materials, Reg. Guide 1.99 (Reference 2) permits calculation of the adjusted reference temperature (ART) using Regulatory Position 1.1. Guidance on which final value to select are given below.

- If Regulatory Position 2.1 gives a higher value of ART than that given by using procedures of Regulatory Position 1.1, the surveillance data should be used.
- If Regulatory Position 2.1 gives a lower value, either may be used.

7.2.2.1 Determine Chemistry Factor (CF)

From Table 1 of Reference 1, the best estimate Cu and Ni weight percent for SA-847 (Heat Number 61782) are:

$$Cu = 0.25\ \%$$
$$Ni = 0.56\ \% \qquad (15)$$

The chemistry factor is interpolated utilizing data given in Table 1 of Reference 2, and the above Cu and Ni values. Hence,

$$CF = 148 + (176 - 148) \times (0.56 - 0.40) / (0.60 - 0.40)$$
$$= 170.4^\circ F \qquad (16)$$

7.2.2.2 Calculate the Adjusted Reference Temperature

Values of the CF in (16), and the fluence, f, in (11) are substituted into Equation (3) to give the adjustment in reference temperature caused by irradiation.

$$\Delta RT_{NDT} = 170.4 \times 5.01^{(0.28 - 0.10 \times \log 5.01)} = 239.03^\circ F \qquad (17)$$

The adjusted reference temperature is calculated using Equation (1),

$$ART = -4.8 + 239.03 + 48.3 = 282.53^\circ F \qquad (18)$$

7.2.3 Select Value of ART for SA-847

Since Regulatory Position 2.1 gives a lower value of adjusted reference temperature for SA-847, this will be selected per Reg. Guide 1.99 (Reference 2) guideline. Hence,

$$ART_{SA-847} = 270.6^\circ F \qquad (19)$$

Since this value is also less than 300° F, **this weld is NOT a concern for pressurized thermal shock events during the period of extended operation for Ginna.**

8.0 Results

Primary results of this design analysis are summarized in Table 8.1 shown below.

Table 8.1
Adjusted Reference Temperatures for Ginna Beltline Welds During Period of Extended Operation

Beltline Welds	ART Reg. Position 1.1, °F	ART Reg. Position 2.1, °F	Selected ART °F	10CFR50.61 PTS Criterion for ART, °F	Comments
SA-1101	268.18[1]	N/A-no surveillance data available.	268.18[1]	300	Not a PTS concern
SA-847	282.53	270.6	270.6[2]	300	Not a PTS concern

Note:

(1) Based on conservative assumption of Cu and Ni contents of 0.35% and 1.0% respectively, per Reg. Guide 1.99 (Reference 2) and an assumed fluence at the weld location of 3.96×10^{18} n/cm^2, E>1Mev, which is twice the predicted value for 54 EFPY (Reference 1).

(2) Lower value was selected per guideline in Reg. Guide 1.99 (Reference 2).

ATTACHMENT 5
CLARIFICATION FOR RAI 4.3.5-1
STRUCTURAL INTEGRITY ASSOCIATES REPORT, APRIL 26, 1989

9.7

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April 26, 1989
JFC-89-034
SIR-89-026, Rev. 0

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Subject: ASME Code Section XI Acceptability of the "B" Inlet Nozzle Flaw Indication in the R.E. Ginna Reactor Vessel, Based on Spring 1989 Inservice Inspection Results

Dear Mike:

The subject inservice inspection (ISI) flaw indication has been evaluated by us as acceptable in accordance with ASME Section XI for continued service without repair, as shown on the attached calculation package sheets. Since the flaw, interpreted as an original construction slag defect at approximately midwall of the nozzle-to-vessel weld, is shown by the present UT examination to be smaller than when it was evaluated as acceptable by Teledyne in 1979, that earlier report conservatively bounds the current flaw evaluation.

In summary, our attached flaw evaluation supports the following conclusions:

1. Irradiation effects from the core are negligible at the flaw location,

2. The applied fracture mechanics K for the embedded flaw with a through-wall dimension of 0.48 inches and a length of 4.94 inches is calculated as 7351 psi.$\sqrt{\text{in.}}$ due to the pressure loading and weld residual stresses described in the Teledyne report,

3. The above K provides a margin of 27.2 against an upper shelf reference K (K_{IR}) of 200,000 psi.$\sqrt{\text{in.}}$, compared to a Section XI required margin of 3.16, and

4. Predicted fatigue crack growth, verified by the ISI experience, is negligible.

Please let me know if you require further information.

Very truly yours,

Fred Copeland

J.F. Copeland
Associate

Reviewed by:

S. S. Tang

/mc
attachment

cc: John F. Smith

ATTACHMENT 6

STRUCTURAL INTEGRITY ANALYSIS

RGE-040

CALCULATION PACKAGE

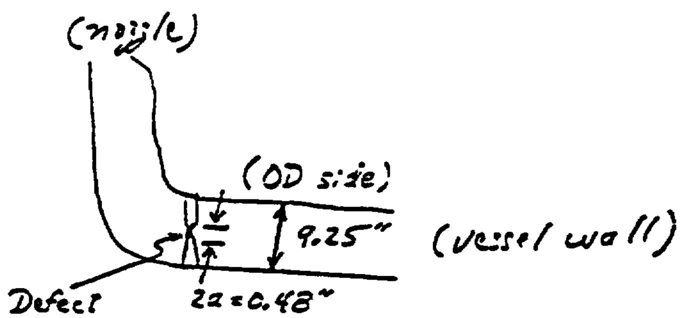
GINNA B INLET NOZZLE ISI INDICATION

OBJECTIVE:

Evaluate the subject indication for acceptability per ASME Section XI [1].

FLAW SIZE AND LOCATION: [2-5]

Embedded construction defect (slag). The location in the nozzle weld is shown [2-4] below and in the attached CAD drawing [5].



ℓ = 4.94" (into paper)
t = 9.25" (at indication location)
e < = 0.625" (on OD side of mid-wall)

$a/\ell = 0.24/4.94 = 0.049$
$2a/t = 0.48/9.25 = 0.052$
$2e/t = 1.25/9.25 = 0.135$

STRESSES:

From the 1979 Teledyne report [6],

σ_m = 6,733 psi
$\sigma_b = \sigma_{residual}$ = 8,000 psi.

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K CALCULATION:

See attached App. A. (Sct. XI) sheets for applicable curves.

$$K_I = \sigma_m M_m \sqrt{\pi a/Q} + \sigma_b M_b \sqrt{\pi a/Q}$$

$$= (\sigma_m M_m + \sigma_b M_b) \sqrt{\pi a/Q}$$

Conservatively take σ_{ys} = 42 ksi, as in Teledyne report [6]

$$\frac{\sigma_m + \sigma_b}{\sigma_{ys}} = \frac{6,733 + 8,000}{42,000} = 0.35$$

For the above value and a/ℓ = 0.049, from Figure A-3300-1,

$Q = 1.02$

From Figure A-3300-2, for 2a/t = 0.052 and 2e/t = 0.135,

$M_m = 1.02$

From Figure A-3300-4, for the same flaw dimensions,

$M_b = 0.21$

$$\therefore K_I = [(6,733)(1.02) + (8,000)\ (0.21)] \sqrt{\pi \frac{(0.24)}{(1.02)}}$$

$$= [6,868 + 1,680]\ (0.86)$$

$$= \boxed{7351 \text{ psi } \sqrt{\text{in.}}} \quad \text{(Applied value)}$$

MATERIAL K (K_{IR}):

From WCAP-8503 [7], the outlet (and inlet) nozzles are located ~25" above the top level of the core assembly. Also in that document, the % of peak fluence at that location is about 2%. Since the ISI indication is at the 10:30 location:

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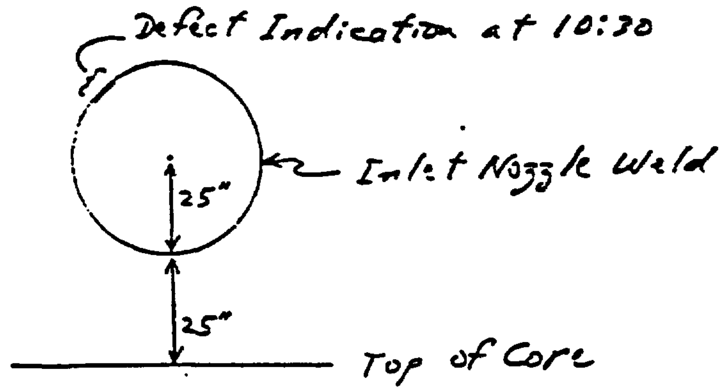

and the radius from the nozzle centerline to the defect location is 25" (Teledyne report), at least an additional 25" can be added to the above 25" number to place the defect at least 50" above the top of the core. It was verified [3] that the defect is, in fact, 57" above the core assembly. From Figure 2-3 (attached) [8], it can be seen that this gives a multiplying factor of less than 10^{-3} times the peak fluence. From the latest Ginna surveillance report (WCAP-10086) [8], the peak measured fluence at the vessel inner surface is 4.03×10^{19} n/cm^2 for 32 EFPYs. Thus, the End-of-Life fluence at the defect location is conservatively established as:

$$(4.03 \times 10^{19} \text{ n/cm}^2) \times 10^{-3}$$

$$= \boxed{4.03 \times 10^{16} \text{ n/cm}^2}$$

That value of fluence is below the threshold for consideration of degradation of toughness by irradiation damage, in accordance with 10CFR50, App. H. (No surveillance, etc. is required for locations with EOL fluence less than 10^{17} n/cm^2. Note that the ISI defect is at about mid-wall, and would see even less fluence.

Thus, the upper shelf K_{IR} value of 200 ksi $\sqrt{in.}$ used in the 1979 Teledyne report and in WCAP-8503 is still appropriate, since the beltline P-T limits assure that the inlet nozzle will be on the upper shelf, as stated in the Teledyne report.

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$$\boxed{K_{IR} = 200{,}000 \text{ psi } \sqrt{\text{in}}}$$

for the inlet nozzle

FATIGUE CRACK GROWTH

The fatigue crack growth law for subsurface cracks, from ASME Section XI, is:

$$\frac{da}{dN} = 0.0267 \times 10^{-9} \quad \Delta K^{3.726}$$

where da/dN is in./cycles and ΔK is in ksi $\sqrt{\text{in.}}$

From prior calculations in this package, the ΔK_I due to going from 0 to 2500 psig is:

$$K_{I,m} = \Delta K = 0.86\ (6{,}868)$$
$$= 5907 \text{ psi } \sqrt{\text{in.}}$$

Substituting this ΔK into an equation to account for mean stress due to the residual stress gives:

$$K_{effective} = \Delta K/(1-R)^m$$

where: $m = 0.5$
$R = K_{min}/K_{max}$
$= 1444/7351$
$= 0.2$

$$\therefore K_{effective} = 5907/(1-0.2)^{0.5}$$

$$= 6604 \text{ psi } \sqrt{\text{in.}} = 6.604 \text{ ksi } \sqrt{\text{in.}}$$

Substituting $K_{effective}$ into the da/dN law to gain an estimate of crack growth rate gives:

$$da/dN = 2.67 \times 10^{-11} (K_{effective})^{3.726}$$

$$= 3.03 \times 10^{-8} \text{ in/cycle}$$

Even assuming 1200 full pressure cycles (0 to 2500 psig) in the 40 year life of the plant (30/yr.), which is conservative, as shown on the attached tables of transients [7,9], the predicted crack growth for 1200 cycles is insignificant:

$$\Delta a = (1200)(3.03 \times 10^{-8})$$

$$\boxed{3.6 \times 10^{-5} \text{ in.}}$$

The above value is not enough to change the value of ΔK and the crack growth rate is relatively constant and insignificant.

As mentioned in the Teledyne report [6], thermal stresses at this mid-wall location are expected to be insignificant.

CODE SAFETY FACTORS:

The Code (Sct. XI) requires a safety factor of

$$\frac{K_{IR}}{K_I} = \sqrt{10} = 3.16$$

The actual safety factor in this case is

$$\frac{K_{IR}}{K_I} = \frac{200{,}000}{7{,}351} = \boxed{27.2}$$

CONCLUSION:

The subject ISI indication is acceptable in accordance with ASME Section XI. No repair is necessary. Since the indication is currently shown as smaller in 1989 than it was in 1979, the 1979 analysis and report submitted to the NRC conservatively envelopes the evaluation of this indication.

REFERENCES:

1. ASME Code, Section XI, 1983 edition or 1986 edition.
2. Telecopy, M. Saporito (RG&E) to J. F. Copeland (SI), 4-6-89.
3. Letter J. F. Smith (RG&E) to J. F. Copeland (SI), 4-11-89.
4. Letter, J. F. Smith (RG&E) to J. F. Copeland (SI), 4-12-89
5. CAD Drawing of Ginna Inlet N2B Nozzle Weld Showing ISI Indication Location, J. F. Smith (RG&E) to J. F. Copeland (SI), 4-23-89.
6. "ASME Section XI Fracture Mechanics Evaluation of Inlet Nozzle Inservice Inspection Indication," Teledyne Technical Report No. TR-3454-1, R.E. Ginna Unit No. 1 Reactor Vessel, March 15, 1979.
7. W. K. Ma, "ASME III, Appendix G Analysis of the Rochester Gas & Electric Corporation, R. E. Ginna Unit No. 1 Reactor Vessel", Westinghouse WCAP-8503, July, 1975.
8. S. E. Yanichko, et al, "Analysis of Capsule T from the Rochester Gas and Electric Corporation R. E. Ginna Nuclear Plant Reactor Vessel Radiation Surveillance Program", Westinghouse WCAP-10086, April 1982.
9. "Thermal Transients and Categories," Ginna Nuclear Power Plant, Appendix H, RG&E, July 15, 1975.

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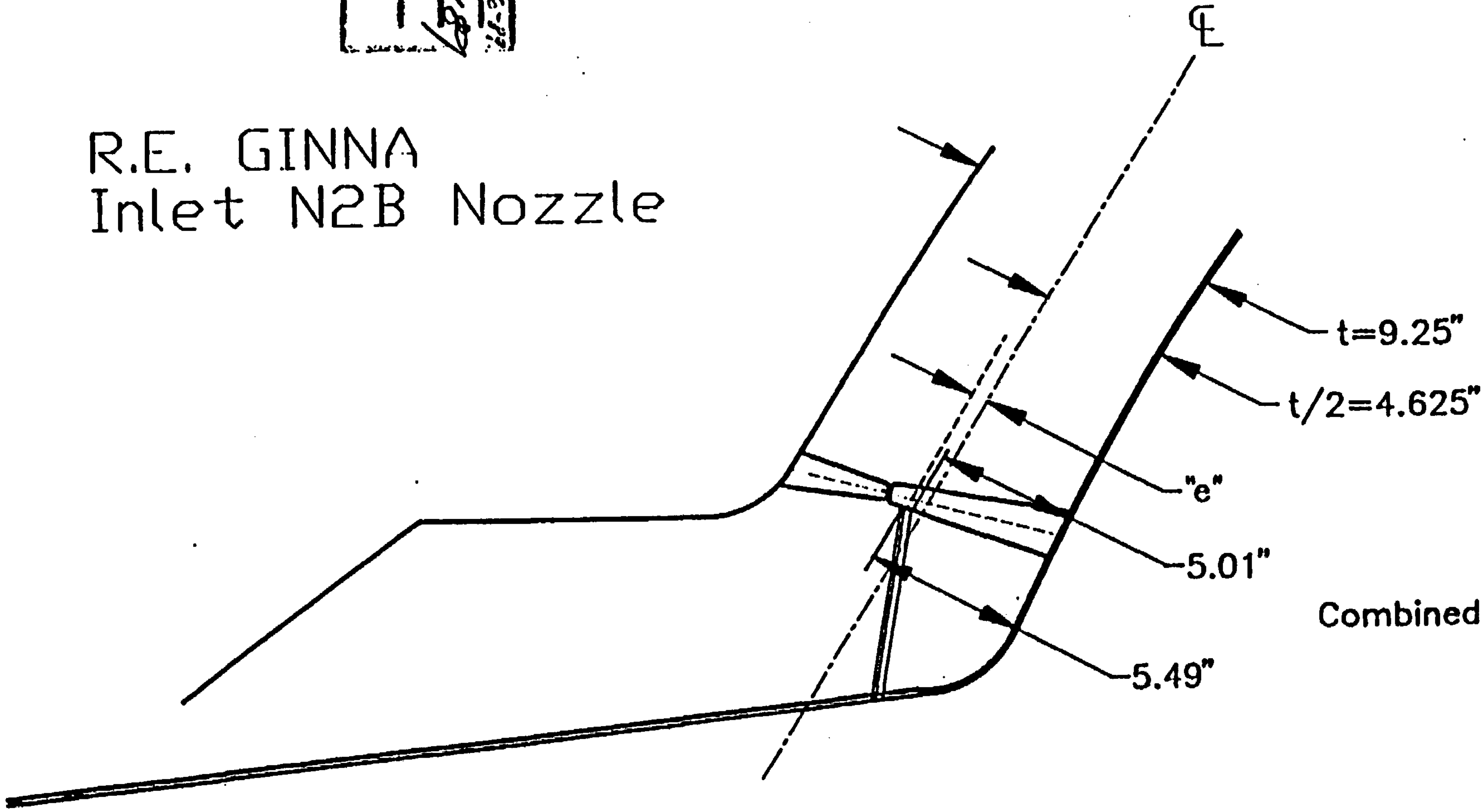
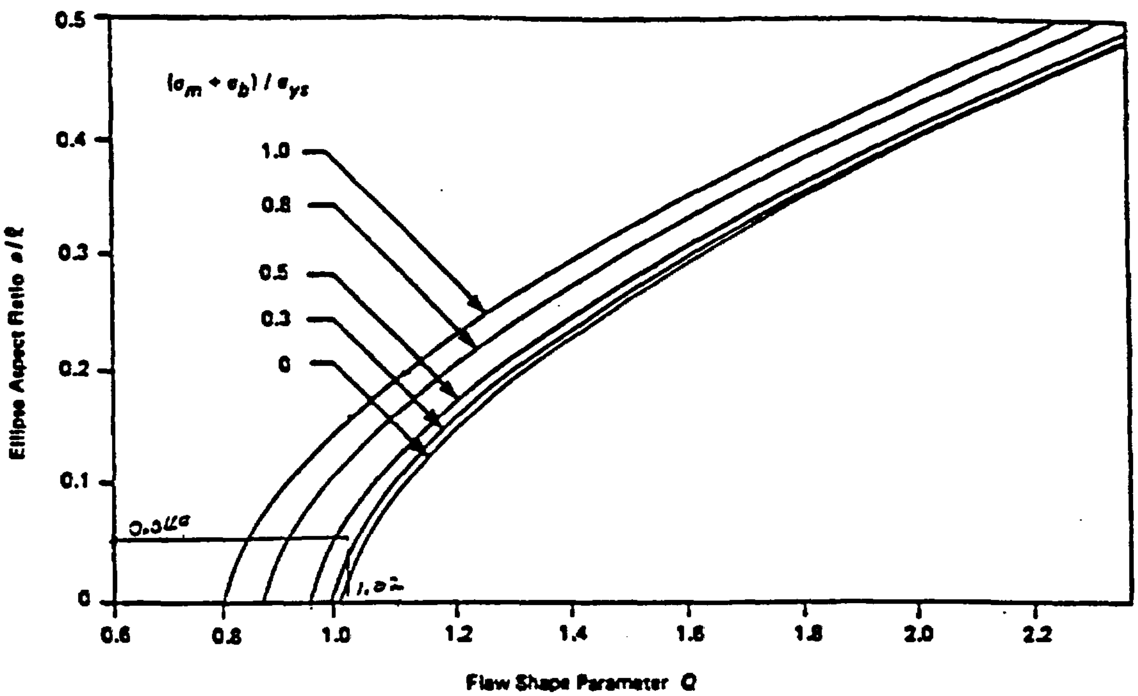
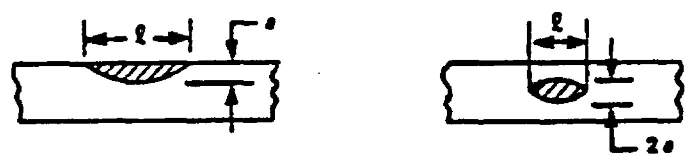
R.E. GINNA
Inlet N2B Nozzle







(a) Surface Flaw (b) Subsurface Flaw

σ_{ys} = specified minimum yield strength
ℓ = major axis of ellipse circumscribing the flaw

FIG. A-3300-1 SHAPE FACTORS FOR FLAW MODEL

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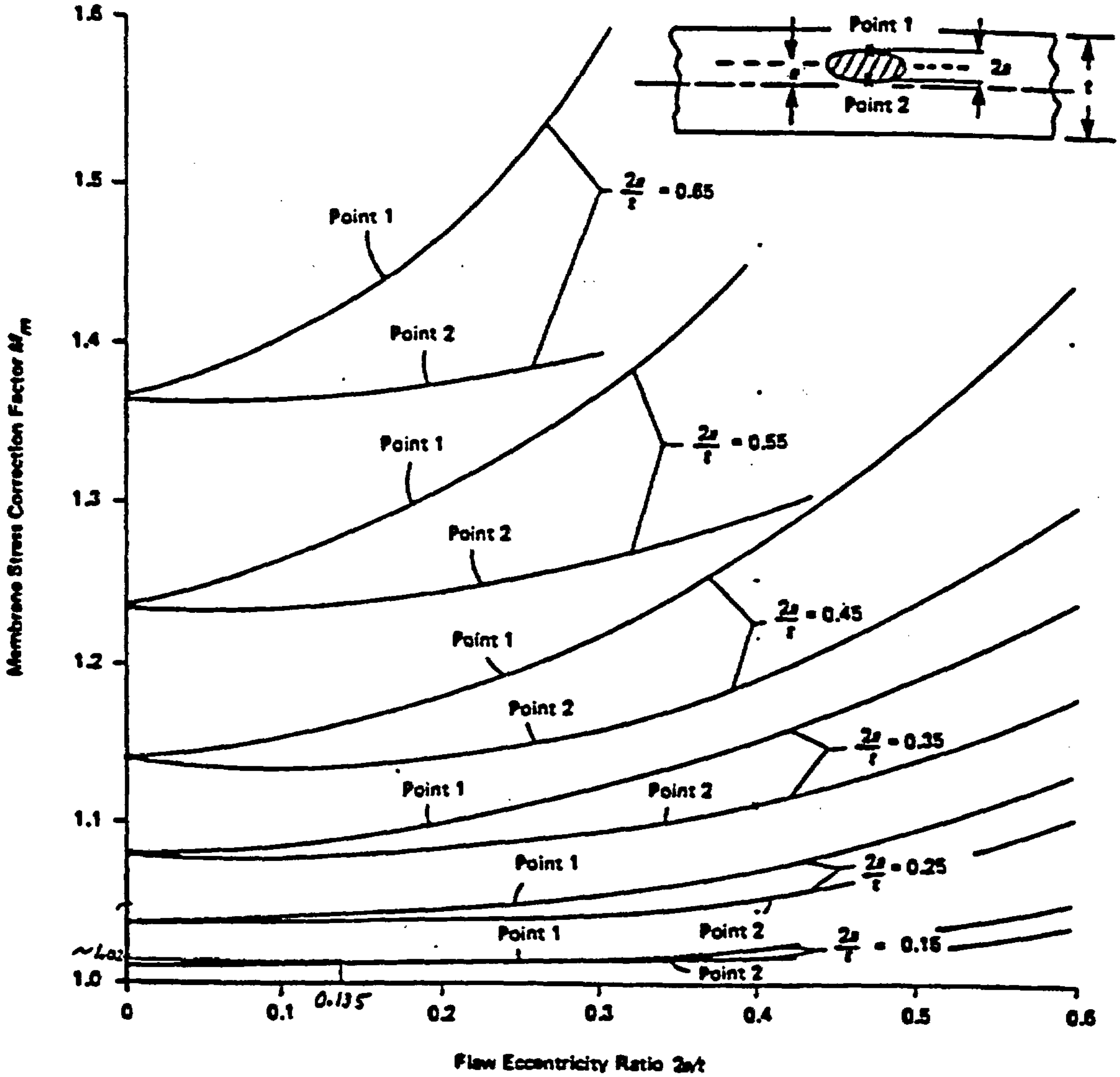

t = wall thickness

e = eccentricity

Point 1 = outer extreme of the minor diameter of ellipse (closer to surface)

Point 2 = inner extreme of the minor diameter of ellipse (further from surface)

FIG. A-3300-2 MEMBRANE STRESS CORRECTION FACTOR FOR SUBSURFACE FLAWS

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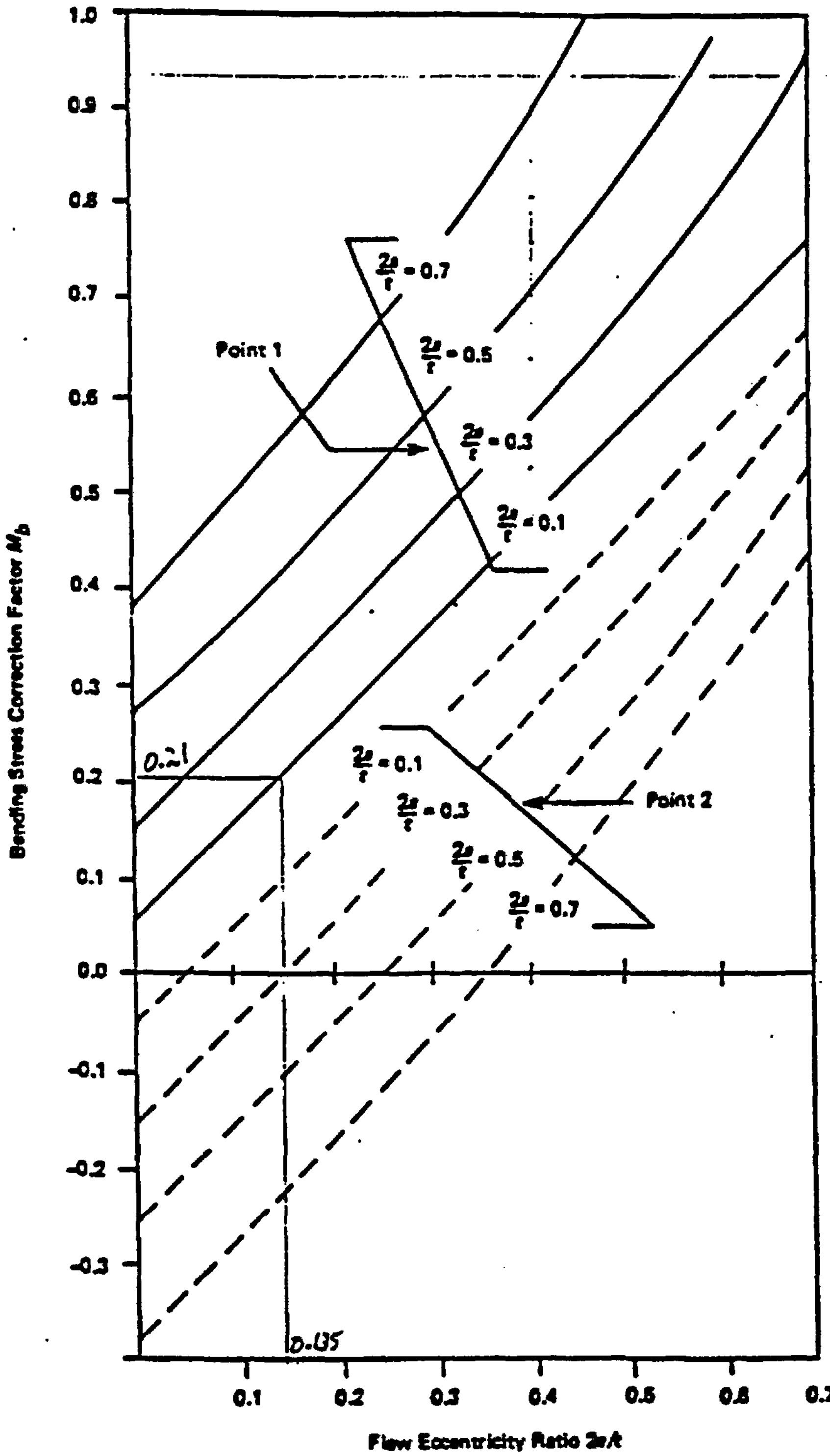

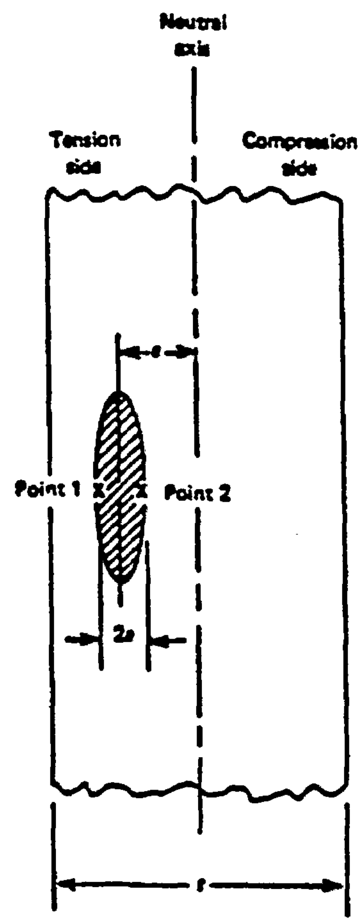

GENERAL NOTE:
If the flaw center line is on the compressive side of the neutral axis, the sign of e_b should be negative.

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FIG. A-3300-4 BENDING STRESS CORRECTION FACTOR FOR SUBSURFACE FLAWS
(For Definitions of Nomenclature, See Fig. A-3300-2)

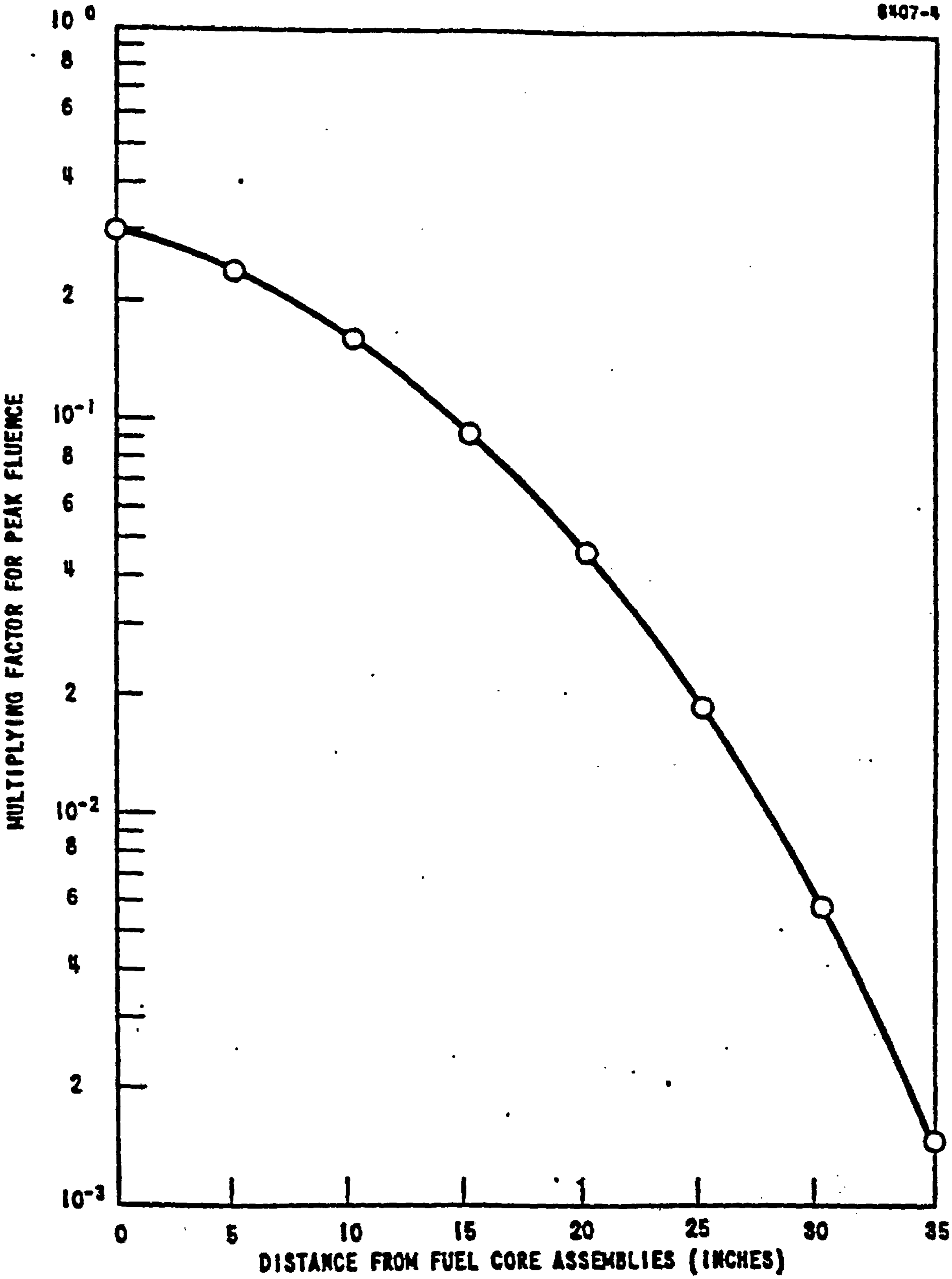

Figure 2-3. Distance Versus Multiplying Factor for Peak Fluence [8]

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TABLE 2-8
TRANSIENTS VS TEMPERATURES [7]

TRANSIENTS	COLD LEG TEMP RANGE FOR CLOSURE HD, BELTLINE, LOWER HD		HOT LEG TEMP RANGE FOR OUTLET NOZZLE	
	LOW (1) (°F)	HIGH (°F)	LOW (1) (°F)	HIGH (°F)
Heatup (2) - Cooldown	70	547	70	547
Plant Loading & Unloading	541	547	547	607
Small Step Load Decrease	539	555	599	612
Small Step Load Increase	527	543	599	615
Large Step Load Decrease	529	554	528	612
Loss of Load	541	575	544	633
Loss of Power	539	553	583	627
Loss of Flow	497	541	492	613
Reactor Trip From Full Power	531	544	529	607
Turbine Roll	475	550	475	550
Steady State Fluctuations	538	544	604	610
Cold Hydro (2)	70	70	70	70
Hot Hydro (2)	50	400	50	400

NOTE (1) : Use the lower temperature for K_{IR} curve comparison; higher limits are for reference only.

NOTE (2) : These transients are structured to ensure compliance with Appendix G.

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TABLE 2-9

TRANSIENTS CONSIDERED IN SUBCRITICAL CRACK GROWTH RATE ANALYSES FOR PRESSURIZER SURGE AND ACCUMULATOR LINES (REFERENCE ~~24~~ -[9])

	Operating Cycle	Occurrences in 40 yr. Design Life
1.	Startup and Shutdown	200
2.	Large Step Decrease in Load (with steam dump)	200
3.	Loss of Load (without immediate turbine or reactor trip)	80
4.	Loss of Power (blockout with natural circulation in Reactor Coolant System)	40
5.	Loss of Flow (partial loss of flow, one pump only)	80
6.	Reactor Trip from Full Power	400
7.	Hydrostatic Test (before initial startup, and post operation)	55
8.	High Head Safety Injection	50
		1105

Assume 1200 Significant Cycles in 40 yr. Design Life (30 cycles/yr.)

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