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July 16, 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: RAI Response Clarifications
R.E. Ginna Nuclear Power Plant
Docket No. 50-244

Dear Mr. Arrighi:

The majority of clarifications were provided by letter dated July 11, 2003. Attachment 1 provides the remaining clarifications. Attachment 1 also provides clarifications and additions resulting from the NRC program audit, conducted June 23-25, 2003.

A regression analysis plot supporting the response to C-RAI 4.5-1 is provided as Attachment 2. A design analysis supporting the response to C-RAI 4.7.4-1 is provided as Attachment 3.

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 July 16, 2003

Robert C. Mecredy

Attachments

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List of Regulatory Commitments

The following table identifies those actions committed to by Rochester Gas & Electric (RG&E) in this document and our previous July 11, 2003 RAI response clarifications. Any other statements in the submittals are provided for information purposes and are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. George Wrobel, License Renewal Project Manager at (585) 771-3535.

Regulatory Commitment		Due Date
C-RAI 3.7-6(a) 7/11/03 letter	Thermographic inspections of 34.5 KV transformer yard components are to be performed at least once per refueling cycle while the components are energized.	Prior to 9/2009
C-RAI B2.1.3-3 7/11/03 letter	Perform visual inspections and UT thickness measurements of the containment liner during 2005 RFO	2005 RFO
C-RAI B2.1.29 7/11/03 letter	Perform hardness tests, if feasible, on EDG jacket water coolers and lube oil coolers channel heads	2003 RFO
C-RAI 3.7-5 7/16/03 letter	Perform visual inspections of phase bus	Prior to 2012

Attachment 1

Clarifications

1. <u>C-RAI 3.7-5</u>

The response to F-RAI 3.7-5 addresses the failures of phase bus identified in the NRC Information Notices referenced in the Staff RAI. The response indicates that the subject phase bus at Ginna are not likely to experience the early failures identified in the Information Notices. It's not clear from the information provided, however, that the phase bus will not be subject to age related degradation over the longer term, 60 year period of plant operation. Several recent applicants have identified the need for aging management programs for phase bus, and a phase bus manufacturer was reluctant to endorse the view that no aging management program is needed for this component. Describe the construction of the phase bus at Ginna, and identify the materials used in their construction. (We note that Table 3.7-2 in the Ginna LRA did not identify the Penetrox (or equivalent) used as an anti-oxidation material in the phase bus, and the AERMs column in the table identifies aging effects for organics but no organics are identified in the material column.) Provide a discussion of how the construction will preclude the ingress of dust, contamination or moisture over a 60 year period. Provide details of the AMR conducted for the phase bus insulating materials, the Penetrox (or equivalent), and any organics used in construction. Include the pertinent parameters and results of the aging analysis which found these materials were good for sixty years. Describe how the AMR determined that connection surface oxidation of aluminum bus connections was not a problem and required no aging management. Provide the phase bus manufacturer's endorsement of your conclusion that thermal cycling of the bus connections at Ginna is not considered significant, and does not require any type of screening such as torque checks or thermal scanning over a 60 year period.

<u>Response</u>

The Ginna Station methodology for performing an Aging Management Review is consistent with NEI 95-10, NUREG-1800, and NUREG-1801. It is noted that the guidance in NEI 95-10 has been found to be acceptable by the NRC staff. The implementation of an aging management program at another nuclear power plant by itself does not determine that it will be applicable to all license renewal applicants. Reviews of recent applicant responses to staff inquiries for aging effects related to phase bus indicate that the applicants did not identify aging effects for phase bus, regardless of whether an aging management program was ultimately adopted. NEI 95-10 supports a plant specific review, while considering industry operating experience. The Ginna Station methodology for performing an Aging Management Review relies upon knowledge of the materials of construction, environment, stressors, and operating experience to the extent that this information is available. During the process of confirming information for this response, apparent discrepancies between analyzed and as-built conditions have been identified. In addition, clarification of original AMR information is provided. The discrepancy resolutions and clarifying information provided in this response will be included in a subsequent AMR revision.

<u>Description of Phase Bus</u>

The phase bus at Ginna within the scope of license renewal is used to provide offsite power from the station auxiliary transformers to Bus 12A and Bus 12B. All phase bus discussed is non-segregated phase bus. The outdoor phase bus installed in the transformer yard was replaced in 1989 to support an offsite power reconfiguration. The outdoor phase bus (Unibus) contains copper conductors and uses an aluminum enclosure. It is non-ventilated, however it contains screened breathers on the bottom of the bus enclosure and electric space heaters for moisture control. Covers are sloped to shed water and gasketed to assist with water tightness. The design of the Unibus provides for overhanging metallic channels such that the gasket is not challenged by normal precipitation. This portion of the switchyard was installed in 1989 and will have 40 years of operation at the end of the period of extended operation.

The indoor phase bus (Westinghouse bus) is original plant construction and begins at the "link" separating the transformer yard from the control building identified on drawing LR33013-2409.

This phase bus is constructed of aluminum conductors and uses a steel enclosure. This phase bus is non-ventilated and does not contain vents, breather screens, or electric space heaters.

Aging Effects for Phase Bus

Only those aging effects that have been experienced by nuclear power plants or discussed as credible in industry reports are considered in this aging management review. As identified in industry operating experience, phase bus is subject to age related failure in the presence of moisture, contaminants, and insulation failure. Some isolated operating experience has identified an aging effect related to highly loaded busses that may experience thermal effects at connections. An aging effect possibly related to poor design practice indicates that Noryl insulation may be subject to cracking in the presence of a chemical interaction with GE D50H47 bus bar compound. These effects have been identified in Information Notices 89-64 and 2000-14. A review of plant specific operating experience did not identify any aging effects for phase bus. Therefore only those aging effects identified in IN 89-64, IN 2000-14, and other industry operating experience will be considered.

Ginna Station performed a visual inspection of both the Unibus and the Westinghouse bus in 2002. This inspection confirmed a lack of moisture, significant contaminants, and insulation degradation. As implied in IN 89-64, all three stressors are present during bus failure. The inspections of the phase bus were conducted by both maintenance personnel and engineers familiar with insulation aging degradation. Digital photographs were taken of both bus bar insulation, insulated supports, and insulation surrounding splice connections. There is no evidence that the aging effects identified in IN 89-64 are present at Ginna Station, or will manifest themselves within the period of extended operation. Since a material compatibility problem would have manifested itself within the first 30 years of operation, there is reasonable assurance that there will be no material compatibility problems over the next 30 years of operation.

Conductor Heat Rise and Bolting Stress

One possible aging effect is related to the thermal expansion and loosening of bus bar conductor connection hardware. As indicated in IN 2000-14, such a condition is consistent with bus bars that are loaded near capacity or even overloaded. Root Cause #2 states that the normal operating temperature of the 12KV phase bus at Diablo Canyon was routinely loaded at approximately 84% ampacity and worst case operation was 100% of bus ampacity. The rated ampacity for the 4 kV phase bus at Ginna Station is 3000 A. The normal loading of the phase bus within the scope of license renewal is less than 500A under single offsite source configuration (100/0 or 0/100). Since 50/50 operation is common, the normal loading splits the current over the two busses. Under startup conditions these conductors may experience a short term increase in current to carry station auxiliary loads. Such loads may cause an increase of no more than 1250 A. Therefore under worst case loading conditions, the maximum current experienced by the phase bus is conservatively calculated at 1750 A. For additional conservatism, the phase bus ampacity was derated to 2500A for calculation purposes. The calculation results shown below are conservative, but encompass actual temperatures expected.:

Normal Loading = 20% of ampacity

Normal Service Temperature < 45°C

Worst Case Loading = 70% of ampacity

Worst Case Service Temperature < 75°C

Under worst case conditions, the service temperature is significantly less than the 105°C temperature limit identified by the vendor in accordance with ANSI standards. This is in contrast to the temperature rise calculation documented in IN 2000-14 which was 46°C normal and 63°C worst case rise above ambient or a service temperature of 86°C and 103°C respectively. The worst case service temperature at Ginna Station does not exceed the normal service temperature derived from IN 2000-14. Under the temperature conditions analyzed for Ginna Station, plastic deformation of connection hardware will not occur. This is supported by section 7.2.4 of EPRI TR-104213, Bolted Joint Maintenance and Application Guide. This section specifically analyzes torque for electrical bus bars. Based on analysis and industry guidance, there is reasonable assurance that bolt torque relaxation is not an aging effect requiring management at Ginna Station.

It is noted that the installation and maintenance manual for Unibus states,

> "Field experience has verified that retorquing of properly installed conductors connection hardware is not required on a routine basis. Connections that are suspect should be inspected and retorqued as required."

Considering the mild operating conditions for the phase bus within the scope of license renewal, there is no reason to suspect that any connections would need to be inspected or retorqued.

Review of Insulating Materials and Anti-Oxidant

Service temperature for phase bus insulation was calculated to be 56.3°C for the purpose of evaluating thermal life of insulating components. This temperature is reasonable considering the normal current carried by the phase bus. Aging information was not readily available for the exact materials of contruction, however the service temperature was evaluated for all materials identified in EPRI 1003057 table B-3. Materials such as Bus Bar Insulation Tube and Epoxy have 60 year service temperatures of at least 65°C. Other materials such as fiberglass/composites, Silicone Rubber, and EPR all have 60 year service temperatures above 65°C. While the use of PVC tape is for mechanical protection and insulation jacketing, PVC boots may provide protection at many splice connections. Other taping materials used appear to be coated in varnish, which will significantly alter the insulating ability of the base tape material. While varnish was not specifically analyzed in the license renewal AMR, the aging characteristics are available in EQ program references and have no aging effects requiring management within the period of extended operation at the temperatures under consideration. While the original AMR considered all Westinhouse splices to be tape wrapped based on installation instructions, photographs confirm that removable boots are used. It is reasonable and conservative to consider these connections to be constructed of PVC. The use of the varnish taping method may be limited to connections within the switchgear. There is reasonable assurance that all insulating materials, except the PVC boots will perform their design function throughout the period of extended operation. In lieu of a materials analysis, visual inspections of boots installed on the Westinghouse bus is recommended to identify potential degradation due to thermal effects. This inspection will be added to procedures for existing periodic switchgear inspection and preventative maintenance.

Oxidation of phase bus connections is considered for initial installation, and long term life. During initial construction and installation, preparation of connection surfaces is important to assure a low resistance connection. It is common practice to plate bus bar connection surfaces with silver or tin to prevent oxides or insulating films from forming. Consistent with guidance for joining plated connections, "a light layer of a joint compound is acceptable, but not mandatory" (IEEE 141-1993). In the initial response to F-RAI 3.7-5, it was implied that Penetrox or an equivalent joint compound has been used on the connection surfaces. Upon further investigation is was found that this was not required for most connections, and Penetrox cannot be considered equivalent to other joint compounds such as No-Ox. Joint compounds such as Penetrox are not used on plated connections because it may damage the plating material. It is also noted that joint compound was prohibited on all Unibus plated copper connections.

During the offsite power reconfiguration, the Westinghouse bus was cut and a splice box was built to transition to Unibus. It is assumed that Penetrox was used to connect the aluminum to the copper transition piece because the Westinghouse bus was not plated at the field cut/prepared end. In this location, the anti-oxidant material is credited with preventing oxidation of the connecting surfaces. Penetrox contains conductive zinc particles in a grease-like base material. The zinc particles assist with breaking down conductor oxides and forming a conductive path, while the grease assists with application and provides some protection against moisture. Since the bus is operated at low temperatures, adequately protected from the environment, wrapped in EPR insulating tape and PVC jacketing tape, and moisture is controlled by the bus heaters, the environment will not support galvanic corrosion. Also considered is that these connection surfaces were constructed in 1989, and will have 40 years of operation upon the end of the License Renewal period of extended operation.

Under some conditions, oxidation could occur when a connection loses sufficient pressure and/or when excessive temperatures are present. As previously stated, there is reasonable assurance that bolt torque relaxation and high temperature conditions will not occur, and therefore long term oxidation of phase bus connections is not an aging effect requiring management at Ginna Station. No operating experience or industry reports were identified that suggests that long term oxidation of bolted connections will occur once splice connections are joined under sufficient pressure, and loaded to less than 80% of ampacity.

Conclusion

The information provided in this response summarizes the conclusions of the AMR, provides background information, and also provides clarifying information which will be included in a subsequent revision to the AMR. A revised table has been attached that more accurately reflects the revised analysis and conclusions. Due to a lack of data for the boot material used on Westinghouse splice connections, a visual inspection will be added to procedures for switchgear inspection and preventative maintenance. Implementation of an Aging Management Program was determined to be conservative since degradation of the PVC boots under analyzed temperatures would not, by itself, lead to phase bus failure. Switchgear maintenance procedures and requirements for administrative controls will be referenced within the basis document for the Periodic Surveillance and Preventative Maintenance Aging Management Program previous submitted in the LRA and modified by RAI responses. The scope attribute of this program will be modified to indicate that phase bus inspections are included within the program. Since inspections were performed in 2002, inspections will be required to be performed once prior to 2012 and continue consistent with scheduled bus inspections and maintenance.

The 11A and 11B phase bus is subject to more significant stressors than 12A and 12B. While not subject to aging management review, 11A and 11B phase bus were also inspected during 2002. This phase bus has not been replaced and is subject to larger loading and resulting temperatures/stresses. Any failures or aging effects on this phase bus would be a leading indicator that the 12A and 12B phase bus may be subject to the same aging effects at some time in the future. The lack of aging effects on the 11A and 11B phase bus provides evidence that there are no aging effects requiring management. The program owner will be provided with the option of substituting inspections of 11A and 11B phase bus instead of performing inspections of 12A and 12B phase bus.

Electrical and Instrumentation and Control Systems- Component Types Subject to Aging Management no Evaluated in NUREG-1801					
Component Types	Material	Environment	AERMs	Program/Activity	Discussion
(1) Electrical Phase Bus	Aluminum, Copper (bus, solid and flexible connectors and straps), Steel (bolts, washers, nuts, etc.), Rigid Bus Parts (porcelain, fiberglass/composite insulators, etc.) Various Organic Insulation (EPR, SR, PVC, Varnished Tape, Bus Bar Insulation Tube, Epoxy, etc.) Zinc joint compound	Indoor (No Air Conditioning), Outdoor	Thermal/ thermoxidative degradation of PVC splice boots.	Periodic Surveillance and Preventive Maintenance	Based on analysis, there is a potential that the PVC boots surrounding original splice connections may experience heat related degradation. Inspections performed in 2002 indicated that there was no visual signs of degradation after 32 years of operation. The potential aging effect of thermally induced bolt torque relaxation was evaluated. Based on the low loading of this phase bus, the thermally induced stresses were well within the design limits of the connection hardware. Aging effects related to connection surface corrosion were also evaluated and it was determined that there are no aging effects requiring management. This conclusion was based on material preparation, installed environment, low relative temperatures, and a review of operating experience.

2. C-RAI 4.3.7-1(a)

The charging nozzle and safety injection nozzle have relatively low calculated fatigue usage. The calculated usage factors do not correlate very well with the usage factors contained in NUREG/CR-6260. Why is the calculated fatigue usage is so low for these components.

Also, provide the projected fatigue usage factor for the reactor vessel shutdown cooling nozzles (P&ID 33013-1260-LR, location F-4) for 60-years of operation.

Response

Both the charging nozzle and the SI nozzle cumulative fatigue usage factors were calculated using plant-specific transients and geometry. Specifically for the charging nozzle, a recent loss and delayed recovery of charging and letdown transient was evaluated using FatiguePro, with a resulting incremental CUF of 0.000112 at the bounding location in the charging nozzle for that one cycle. This transient is the bounding transient for the charging nozzle. The peak stress range was 115.8 ksi and the K_e was computed as 1.089. 300 cycles of this transient produces a CUF = 0.0336. The actual stress (strain) history was evaluated using the equations contained in NUREG/CR-5704 to determine an overall F_{en} for the transient, as follows:

The NUREG/CR-5704 methodology prescribes that the F_{en} be integrated over the tensile strain range of the transient being analyzed. Using this methodology, the Fen factor is computed as a function of three factors specified in Ref. [2] using the following equation:

F_{en} = exp (0.935-T* εdot* O*)

where:

T* = 0	T < 200C
T* = 1	T $\geq$ 200C
εdot * = 0	εdot > 0.4%/sec
εdot * = ln (εdot/0.4)	0.0004 $\leq$ εdot $\leq$
0.4%/sec	
εdot * = ln (0.0004/0.4)	εdot < 0.0004%/sec
O * = 0.260	DO < 0.05 ppm
O * = 0.172	DO $\geq$ 0.05 ppm

For this transient, the factors :

Strain range - over the tensile portions of the plant transient
Strain rate - computed over the tensile strain range
Temperature - conservatively assumed to be greater than 200°C
Dissolved oxygen - conservatively assumed to be less than 0.05 ppm

The F_{en} thus evaluated was determined to be 7.56 for this transient. Thus, the environmentally-assisted fatigue usage for the design number of cycles (300 transients) is:

CUF = 0.0336 x 7.56 = 0.254

If we had used a conservative bounding F_{en} factor of 15.35, the environmentally-assisted fatigue usage value would have been:

CUF = 0.0224 x 15.35 = 0.516

These values compare to an environmentally-assisted fatigue usage value for the charging nozzle in NUREG/CR-6260 Section 5.4 (Table 5-92) of 0.319.

For the safety injection nozzle, NB-3600 piping analysis was performed. This analysis was performed for the design transients and cycles. The resulting fatigue usage is 0.0164. Applying an environmental factor of 15.35 yields an environmentally-assisted fatigue usage of 0.252.

This value compares to an environmentally-assisted fatigue usage value for the safety injection nozzle in NUREG/CR-6260 Section 5.5 (Table 5-95) of 0.327.

The usage factor for the "reactor vessel shutdown cooling nozzles," called the "low head safety injection nozzles" at Ginna Station, was calculated as 0.02 per MPR Associates Report, dated June 30, 1969. This included 10 full-cycle 550°F to 70°F thermal cycles. since the number of cycles projected for 60 years of operation (4) is less than the number of cycles used in this analysis (10), the calculated CUF of 0.02 remains binding for 60 years of operation.

3. <u>C-RAI 4.5-1</u>

In constructing the trend lines, it appears that the analyst has averaged the prestressing forces measured during each inspection. In Information Notice 99-10, the staff discourages the averaging method. The regression analysis is more representative, if each measured value is independently considered in the regression analysis. The individual measured values plotted on both sides of the trend line would make the operating experience transparent. the applicant is requested to show the individual measured values obtained during each inspection.

<u>Response</u>

A regression analysis, including the individual measured values obtained during each lift-ff test, we provide as Attachment 2.

4. <u>C-RAI 4.7.4-1</u>

The applicant is requested to provide the bases for (1) 0.030 in (vertical), (2) 0.0014 (radial) displacements, and (3) 144 cycles used in the final fatigue usage factor calculations.

<u>Response</u>

The bases for the (1) 0.030 inch (vertical), (2) 0.0014 inch (radial) displacements, and (3) 144 cycles used in the final fatigue usage factor calculations are contained in Attachment 3, Design Analysis DA-CE-202-016-07, Rev. 0, dated 6/22/02.

The following clarifications are being provided in response to the June 23-25 program audit:

1. <u>B2.1.26</u>

The name of this program has been changed to the "Nickel-Alloy Nozzles and Penetrations Inspection Program", consistent with NUREG-1801. All nickel-alloy components (Alloy 600 and Alloy 690) exposed to the reactor coolant system environment determined to have an intended function for license renewal and their associated nickel-alloy attachment welds (Alloy 82/182 and Alloy 52/152) are within the scope of the Nickel-Alloy Nozzles and Penetrations Inspection Program. These components include the CRDM adapter tubes and vent pipes (presently Alloy 600), bottom-mounted instrument penetrations (Alloy 600), and the radial support lugs (Alloy 600). The butt welds joining the nickel-alloy CRDM tube to the stainless steel adapter (presently Alloy 82/182) and the bottom-mounted instrumentation nozzle-to-stainless steel safe ends (Alloy 82/182) are in scope. The weld overlay cladding on the tubesheets of the replacement steam generators (Alloy 82) and the weld buttering (Alloy 152) on the replacement steam generator primary inlet and outlet nozzles are also in scope.

After installation of the new reactor vessel head (scheduled for the Fall 2003 refueling outage), the CRDM adapter tubes will be Alloy 690. The partial penetration attachment welds joining the tubes to the head will be Alloy 52. The butt weld joining the tube to the stainless steel adapter will be Alloy 52. The vent pipes will be SA-312, Type 304L stainless steel joined to the head to Alloy 52 partial penetration attachment welds.

The conclusion of this program description should state that the Nickel-Alloy Nozzles and Penetrations Inspection Program is a new program that is consistent with NUREG-1801 (Generic Aging Lessons Learned (GALL) Report), Section XI.M11, "Nickel-Alloy Nozzles and Penetrations" (Ref. 9.1) with an exception. The NUREG-1801 program is based on the use of a susceptibility model to determine inspection timing. The Ginna Station Nickel-Alloy Nozzles and Penetrations Inspection Program is based on the inspection requirements and frequency mandated in the NRC Order issued on February 22, 2003.

2. <u>B2.1.28</u>

The scope of this program description should be modified to note that the Reactor Vessel Surveillance Program meets the requirements of ASTM E-185-82. The scope of the program should state that it applies to P/T Limit Curves, Upper Shelf Energy, Pressurized Thermal Shock, and LTOP setpoints, consistent with NUREG-1801. The conclusion of this program should state that the program is consistent with NUREG-1801, with exceptions. This program does not require testing of the surveillance specimen scheduled to be removed when the fluence exposure is equivalent to 52EFPY.

3. <u>B2.1.30</u>

The conclusion should state that this program is not consistent with NUREG-1801 (although all 10 elements are properly addressed), since the Ginna program addresses aging management of borated stainless steel, while NUREG-1801 addresses aging management of boraflex.

4. <u>B2.1.31</u>

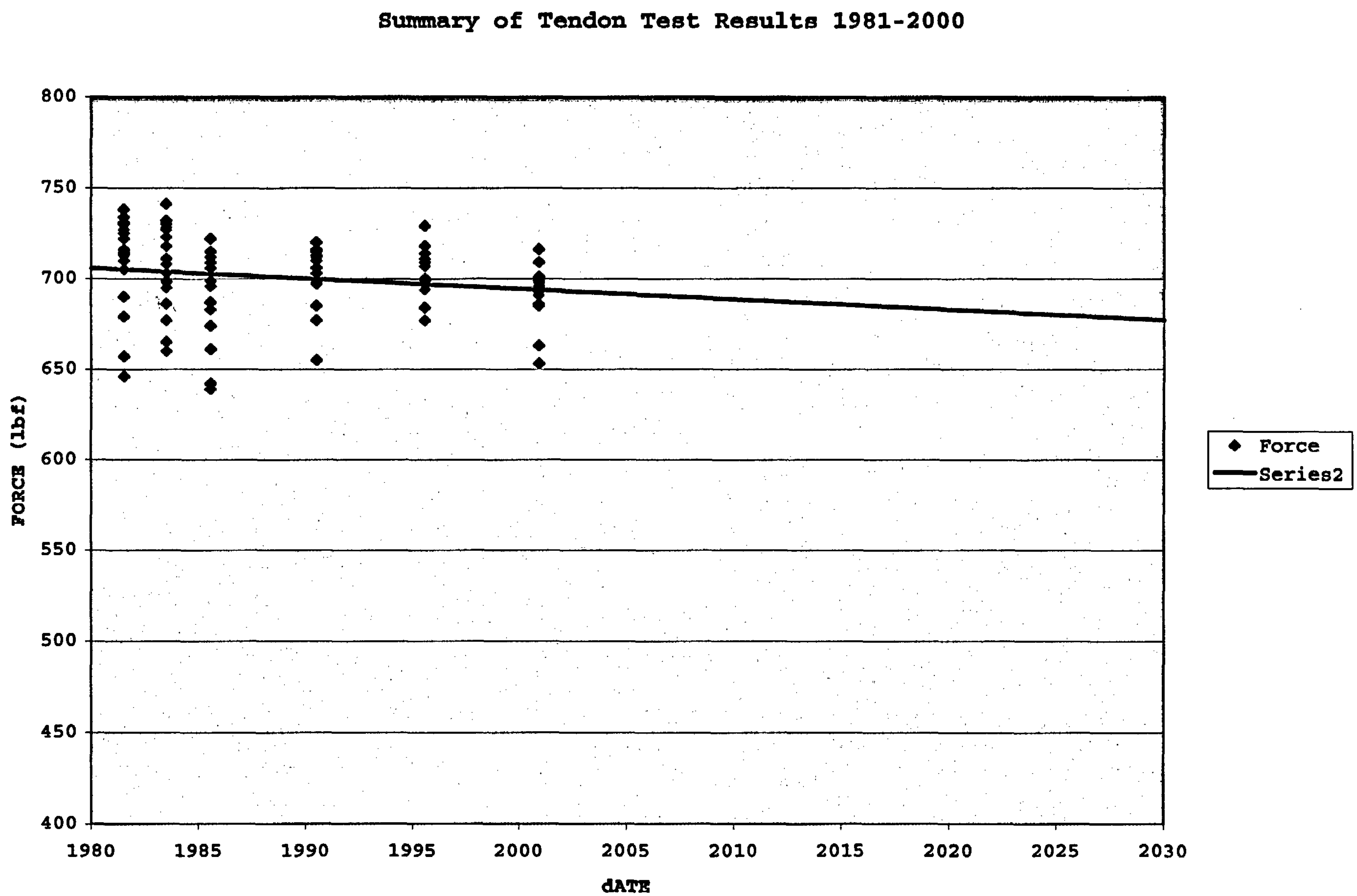
As noted in our July 11 clarification response, the Ginna program manages the effects of the secondary side of the steam generators, as well as the steam generator tubes. The name of the program is being changed to "Steam Generator Integrity Program", and the conclusion at the end of B2.1.31 should state the program is consistent with NUREG-1801, with enhancements, since it manages the aging to the steam generator components beyond those addressed NUREG-1801.

5. <u>B3.1</u>

The conclusion of the program description should have stated the Ginna Station Environmental Qualification Program is consistent with NUREG-1801.

Attachment 2

Summary of Tendon Test Results 1981-2000


Summary of Tendon Test Results 1981-2000
800
750
700
650
600
550
500
450
400
FORCE (lbf)
1980
1985
1990
1995
2000
2005
2010
2015
2020
2025
2030
dATE
Force
Series2

Attachment 3

DA-CE-2002-016-07, Containment Time Limited Agin Analyses for License Renewal: Tendon Bellows Fatigue

D0970547

Containment Time Limited Aging Analyses for License Renewal:

Tendon Bellows Fatigue

Ginna Station

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DA-CE-2002-016-07

Revision 0

Prepared by: G. Bjorkman, June 10, 2002
Assigned Engineer
Gordon Bjorkman
Date

Reviewed by: Len Sucheski 6/22/02
Independent Reviewer
Len Sucheski
Date

Category 4.66.1
Reviewed QA

Revision Status Sheet

Revision Number	Affected Sections	Description of Revision
0	All	Original Issue

Table of Contents

1.0 Purpose

The allowable radial and vertical displacements of the containment stainless steel tendon bellows are given in UFSAR Figure 3.8-18 (Attachment 1) and are limited to two cycles per year for the 40 year life of the plant. This limits the total number of allowable displacement cycles to 80. Assuming that 80 cycles of allowable displacement results in a fatigue usage factor of 1.0, the purpose of this calculation is to demonstrate that the actual and projected cycles and displacements that would occur over a 60 year period result in a fatigue usage factor of less than 1.0.

2.0 Conclusion

The fatigue usage factor of the tendon bellows is calculated to be 0.004 over a 60 year period. Therefore, the structural integrity of the tendon bellows will be maintained through the period of extended operation.

3.0 Design Input

None

4.0 Reference Documents

1. Structural Integrity Test of the Reactor Structure, Gilbert Associates, Inc, GAI Report No. 1720, October 1969.

5.0 Assumptions

As noted in the calculation

6.0 Computer Codes

None

7.0 Analysis

Since the completion of construction, displacements at the tendon bellows have occurred due to pressure testing (Structural Integrity Tests - SITs and Integrated Leak Rate Tests - ILRTs) and temperature changes in the cylindrical shell wall due to summer/winter conditions and reactor shutdown during refueling outages. While these displacements are relatively small, they are nonetheless cycles of displacement that must be addressed to validate the structural integrity of the tendon bellows through the period of extended operation.

SITs were performed in 1969 at 69 psi and 1996 at 72 psi. The 1969 SIT (Reference 1) displacement results were slightly higher than the 1996 SIT results, and therefore, bound the 1996 results. The maximum vertical displacement in Table II of the SIT (Attachment 2) exceeds the vertical displacement design limit of the bellows because it includes the vertical growth of the containment shell between elevations 343'-2" and 231'-8" in addition to bellows expansion. The containment shell vertical expansion, due to internal pressure between these elevations, must be subtracted from the Table II data. The bellows displacement is calculated below.

Calculate the Bellows Maximum Vertical Displacement during the 1969 SIT

From Table II the maximum vertical displacement of the containment cylinder due to internal pressure occurs at 60 psi during depressurization and is equal to 0.175 inches. This displacement is made up of two components; the vertical displacement of the containment concrete due to membrane tension and the vertical displacement of the neoprene pad at the bellows. The displacement of the containment concrete between elevations 343'-2" and 231'-8" can be found from the equation

Delta V = PL/AE,

Where,

L = 343'-2" - 231'-8" = 1338 inches
P = pressure x PI x radius x radius = 60(3.14)(630)(630) = 74,776,000 lbs
A = thickness x PI x diameter = 42(3.14)(1260 + 42) = 171,708 sq-in
E = 4,030,000 psi

Delta V = 0.145 inches

Thus the bellows vertical displacement is 0.175 – 0.145 = 0.030 inches

Determine the Bellows Maximum Radial Displacement during the 1969 SIT

From Table IV (Attachment 2) the maximum radial displacement at elevation 232'-0" is +0.014", which occurred at a pressure of 60 psi during depressurization.

Calculate the Free Radial Thermal Expansion of the Containment Cylinder

Conservatively assume that the change in the average temperature through the cylinder wall between summer and winter is 100 degrees F, and that the change in the average temperature through the cylinder wall between startup and shutdown is also 100 degrees F. Letting,

Delta T = 100 F degrees
Mean Cylinder Radius, R = 630 + 21 = 651"
Alpha of concrete = 6.0E-6 in/in/F degrees

Then the free radial expansion, Delta R is

Delta R = (Delta T)(Alpha)(R) = 100(6.0E-6)(651) = 0.40"

Calculate the Internal Pressure that Produces the Same Radial Expansion

From Table I (Attachment 2) the maximum radial displacement at a depressurization pressure of 60 psi is +0.552 at elevation 288'. Thus the free radial thermal expansion is equivalent to an internal pressure of

(60)(0.40/0.552) = 43.5 psi.

This pressure would produce a radial displacement at the bellows equal to

(43.5/60)(0.014) = 0.010"

Determine Radial and Vertical Displacements for ILRTs

From Attachment 3 the containment pressure during ILRT is 35 psi. The corresponding displacements at the bellows are

Delta R = (35/60)(0.014) = 0.008"
Delta V = (35/60)(0.030) = 0.018"

60-Year Projected Thermal and Pressure Cycles and Displacements

Pressure or Thermal Event	Cycles	Vertical Displ.	Radial Displ.
SITs	4	0.030"	0.014"
ILRTs	20	0.018"	0.008"
Summer/Winter Temp	60	0.0	0.010"
Shutdown/Startup Temp	60	0.0	0.010"

For the purpose of this analysis, assume 144 cycles of maximum vertical and radial displacement equal to 0.030" and 0.014" respectively.

Determine the Bellows Fatigue Usage Factor for 144 Cycles of Maximum Vertical and Radial Displacement

The allowable radial and vertical displacements of the containment stainless steel tendon bellows are given in UFSAR Figure 3.8-18 (Attachment 1) and are limited to two cycles per year for the 40 year life of the plant. This limits the total number of allowable displacement cycles to 80. Assume that 80 cycles of allowable displacement results in a fatigue usage factor of 1.0.

From ASME Code Figure N-415(B) for stainless steel (Attachment 4), the allowable alternating stress amplitude, Sa, for 80 cycles is 260 ksi. Assume that this is the peak stress produced in the bellows due to the combined effect of design basis vertical and radial displacements of 0.10" and 0.16" given in UFSAR Figure 3.8-18 (Attachment 1), and that this stress is directly proportional to the absolute sum of the displacements. Thus, Sa = 260 ksi is produced at an absolute displacement of 0.26".

The absolute magnitude of the combined vertical and radial displacements that actually occur is 0.030" + 0.014" = 0.044". This displacement corresponds to a bellows stress level of

$$(0.044/0.26)(260) = 44 \text{ ksi}$$

The alternating stress intensity amplitude of 44 ksi corresponds to 40,000 cycles. The fatigue usage factor is 144/40,000 = 0.004.

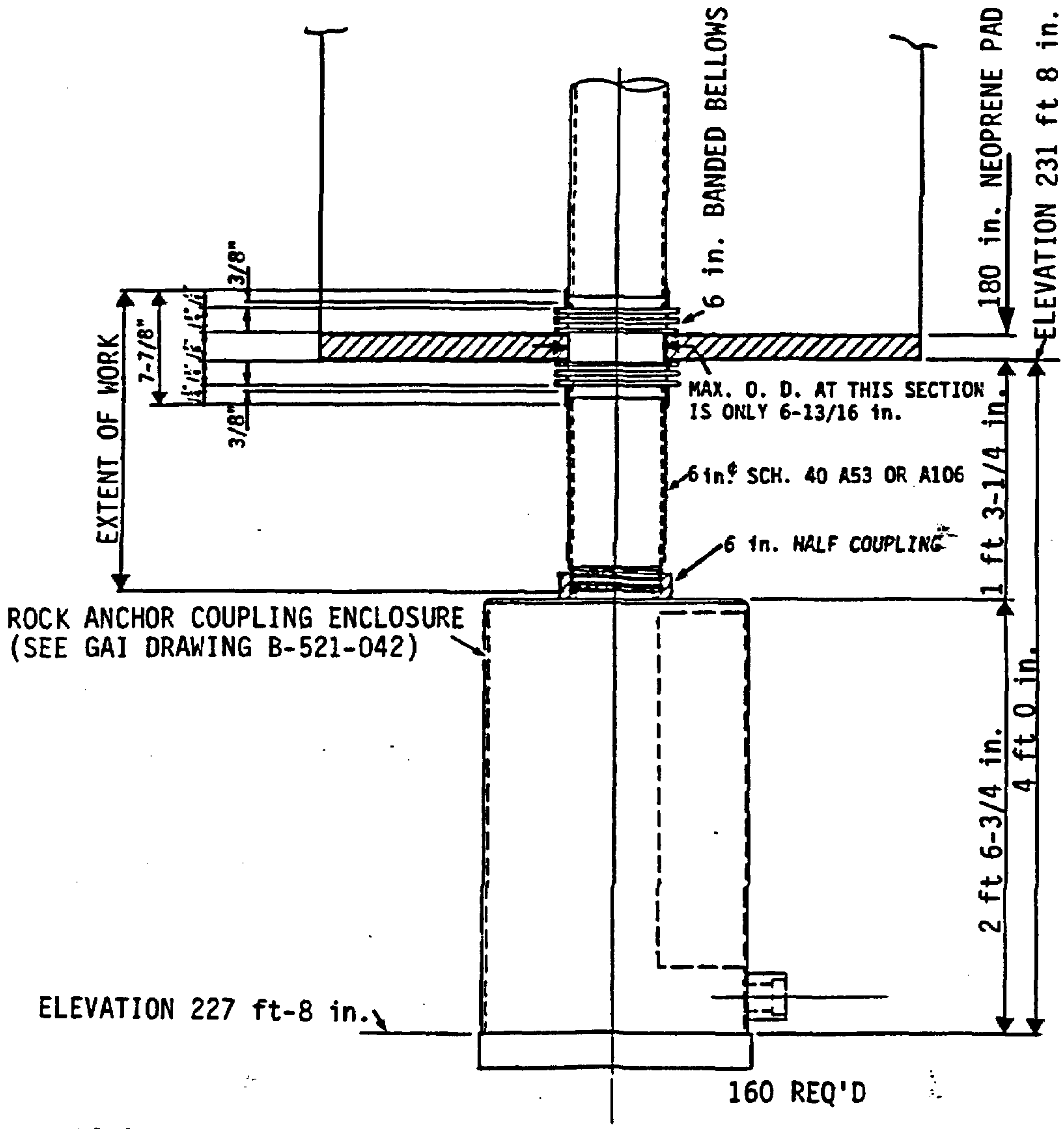
Therefore, the structural integrity of the tendon bellows will be maintained through the period of extended operation.

8.0 Results

The fatigue usage factor of the tendon bellows is calculated to be 0.004 over a 60 year period. Therefore, the structural integrity of the tendon bellows will be maintained through the period of extended operation.

Attachment 1

UFSAR Figure 3.8-18



ENGINEERING DATA

1. MOVEMENTS: CASE (1) FROM UNDEFLECTED POSITION VERTICALLY DOWNWARD 0.14 INCHES.
   CASE (2) FROM ABOVE POSITION VERTICALLY UPWARD 0.10 INCHES AND SIMULTANEOUS LATERALLY 0.16 INCHES.
2. FATIGUE: TWO CYCLES PER YEAR.
3. WORKING PRESSURE: 60 psig.
   TEST PRESSURE: HYDROSTATIC AT 150% OF WORKING PRESSURE.
   PNEUMATIC AT 125% OF WORKING PRESSURE.
4. MAXIMUM WORKING TEMPERATURE: 160°F.
5. STANDARD SPECIFICATION: ASA B31.1 CODE FOR PRESSURE PIPING.
6. TEST TWO RANDOM ASSEMBLIES FOR SPECIFIED MOVEMENTS.

ROCHESTER GAS AND ELECTRIC CORPORATION
R.E. GINNA NUCLEAR POWER PLANT
UPDATED FINAL SAFETY ANALYSIS REPORT

Figure 3.8-18

Containment Miscellaneous Steel Tendon Conduit - Hinge Detail

Attachment 2

Instrument Locations and Tables from Reference 1

2

~~TABLE~~

INSTRUMENT LOCATION
SHEET 1 OF 4

1-1: Cylinder Base Rotation

LVDT Gage Loc.:	Elevation:	Azimuth:	Notes:
1,2,3	232'-0"	100°07'30",220°,340°	Radial Displacement
4,5,6	238'-0"	100°07'30",220°,340°	Radial Displacement
7,8,9	233'-0"	100°07'30",220°,340°	Vertical Displacement

1-2: Cylinder Wall and Dome Displacement - Jig Transit

Scale & Tape Loc.:	Elevation:	Azimuth:	Notes:
10,11,12	260'-0"	40°42',166°07',260°	Radial Displacement
13,14,15	288'-0"	40°42',166°07',260°	Radial Displacement
16,17,18	315'-0"	40°42',166°07',260°	Radial Displacement
19,20,21	337'-0"	40°42',166°07',260°	Radial Displacement
22,23,24	347'-0"	40°42',166°07',260°	Radial Displacement
25,26,27	343'-2"	40°42',166°07',260°	Vertical Displacement

1-3: Equipment Access Opening - Displacement

LVDT Gage Loc.:	Elevation:	Notes:
28,29,30,31,32,33,40	281'-7-1/8"	Radial Displacement
28,29,30,31,32,33,40	281'-7-1/8"	Vertical Displacement
41	272'-5-1/4"	Radial Displacement
41	272'-5-1/4"	Vertical Displacement
34	290'-9"	Radial Displacement
34	290'-9"	Vertical Displacement
35	293'-3"	Radial Displacement
35	293'-3"	Vertical Displacement
36	295'-9"	Radial Displacement
36	295'-9"	Vertical Displacement

TABLE II

VERTICAL DISPLACEMENT DATA

Tape Location	Date	4-11-69	4-11-69	4-12-69	4-13-69	4-13-69	4-14-69	4-14-69	4-15-69	4-16-69	4-17-69	4-18-69
	Time	1728	2400	1050	0831	1604	0010	1140	1745	1008	1720	0700
	Pressure (psi)	0	14	35	50	60	69	60	60	35	35	0
25 (Azim. 40°42')	Inches	0	+0.018	+0.057	+0.099	+0.128	+0.158	+0.143	+0.157	+0.102	+0.109	+0.037
26 (Azim. 166°07')	Inches	0	+0.006	+0.047	+0.086	+0.116	+0.144	+0.128	+0.130	+0.074	+0.074	+0.009
27 (Azim. 260°)	Inches	0	+0.017	+0.057	+0.106	+0.147	+0.173	+0.159	+0.175	+0.117	+0.127	+0.033

NOTE: Data read by sighting a vertical scale using a precision level at the subbasement elevation. The vertical scale was attached to an invar tape suspended from the cylinder wall at Elevation 343'-2".

TABLE IV

HINGE DISPLACEMENT DATA

LVDT Location	Date	4-11-69	4-12-69	4-13-69	4-13-69	4-14-69	4-14-69	4-15-69	4-16-69	4-17-69	4-18-69	4-18-69
	Time	2352	1048	0831	1605	0011	1015	1652	0913	1631	0600	0941
	Pressure (psi)	14	35	50	60	69	60	60	35	35	0	0
1	Displacement (inches)	-0.002	-0.010	-0.009	-0.002	-0.002	0.000	-0.001	-0.012	-0.010	-0.003	-0.003
2		-0.001	-0.005	-0.007	-0.005	-0.001	-0.001	-0.002	-0.004	-0.003	-0.001	0.000
3		+0.001	-0.002	-0.002	+0.007	+0.012	+0.014	+0.012	+0.007	+0.008	+0.006	+0.006
4		+0.002	+0.007	+0.053	+0.087	+0.118	+0.103	+0.101	+0.049	+0.050	+0.012	+0.012
5		+0.002	+0.013	+0.033	+0.052	+0.070	+0.064	+0.062	+0.038	+0.038	+0.012	+0.012
6		+0.005	+0.022	+0.067	+0.093	+0.120	+0.112	+0.110	+0.070	+0.071	+0.023	+0.023
7		+0.002	+0.004	+0.001	-0.001	-0.002	-0.002	-0.004	-0.004	-0.002	-0.001	-0.001
8		+0.001	-0.001	-0.002	-0.002	-0.002	-0.002	-0.004	-0.007	-0.006	-0.006	-0.006
9		+0.001	-0.001	-0.002	-0.004	-0.004	-0.004	-0.006	-0.008	-0.006	-0.004	-0.004

NOTES: 1. Above displacement data were measured using linear variable displacement transducers (LVDT's).
2. Measurements were made relative to reference frames mounted on the ring beam.

TABLE I (CONTINUED)

CYLINDER WALL AND DOME RADIAL DISPLACEMENT DATA

Scale Location	Date	4-17-69	4-17-69	4-17-69	4-17-69	4-17-69	4-17-69	4-17-69	4-17-69	4-17-69	4-17-69	4-18-69
	Time	1920	2345	1200	0904	1700	0015	1140	1800	1010	1655	0645
	Pressure (psi)	0	14	35	50	60	69	60	60	35	35	0
11 (Azim. 166°07') (Elev. 260')	Inches	0	+0.027	+0.155	+0.325	+0.433	+0.545	+0.510	+0.513	+0.333	+0.335	+0.108
14 (Azim. 166°07') (Elev. 288')	Inches	0	+0.027	+0.211	+0.370	+0.472	+0.591	+0.552	+0.549	+0.353	+0.353	+0.120
17 (Azim. 166°07') (Elev. 315')	Inches	0	+0.006	+0.152	+0.270	+0.367	+0.492	+0.466	+0.461	+0.289	+0.292	+0.104
20 (Azim. 166°07') (Elev. 337')	Inches	0	-0.021	+0.048	+0.120	+0.198	+0.321	+0.327	+0.332	+0.227	+0.218	+0.101
23 (Azim. 166°07') (Elev. 347')	Inches	0	+0.056	+0.093	+0.152	+0.227	+0.360	+0.391	+0.395	+0.307	+0.285	+0.202

NOTES:
1. *Data read by vertical sighting of horizontal scales using a jig transit set up over a reference target at the ring beam.* The horizontal scales were attached to the containment wall at five elevations at each azimuth.
2. Data for scales above the spring line have been corrected to indicate displacement in the radial direction.
3. Data for Scale Location 12 are based on an adjusted zero pressure reading. This became necessary because the epoxy cement bond holding the scale bracket broke after the start of the test.
4. The last two readings for Scale Location 24 have been adjusted because the scale was bent by a workman and then straightened.

Attachment 3

Ginna Station SIT/ILRT Testing History

3

Ginna Station- ILRT/SIT Testing History

1969-Containment {CNMT} Structural Integrity Test {SIT}, conducted at 69.0 psig

1969-CNMT Integrated Leak Rate Test {ILRT}, leakage measured at 35.0 psig and 60.0 psig

1972-CNMT ILRT, leakage measured at 35.0 psig

1976-CNMT ILRT, leakage measured at 35.0 psig

1978-CNMT ILRT, leakage measured at 35.0 psig

1982-CNMT ILRT, leakage measured at 35.0 psig

1986-CNMT ILRT, leakage measured at 35.0 psig

1989-CNMT ILRT, leakage measured at 35.0 psig

1993-CNMT ILRT, leakage measured at 35.0 psig

1996-CNMT ILRT, leakage measured at 60.0 psig

1996-CNMT SIT, conducted at 72.22 psig

2006-Current schedule reflects a CNMT ILRT performance, may be extended based on current industry activity whereby a number of licensees have been granted 5 year extensions to the current 10 year baseline interval provided by 10 CFR 50, Appendix J, OPTION B.

Chronological History updated 11/29/01:
Gregg E. Joss, Manager-Tech Spec Program 5.5.15, "CNMT Leakage Rate Testing Program"

Attachment 4

ASME Code Section III-1965 Figure N-415(B)

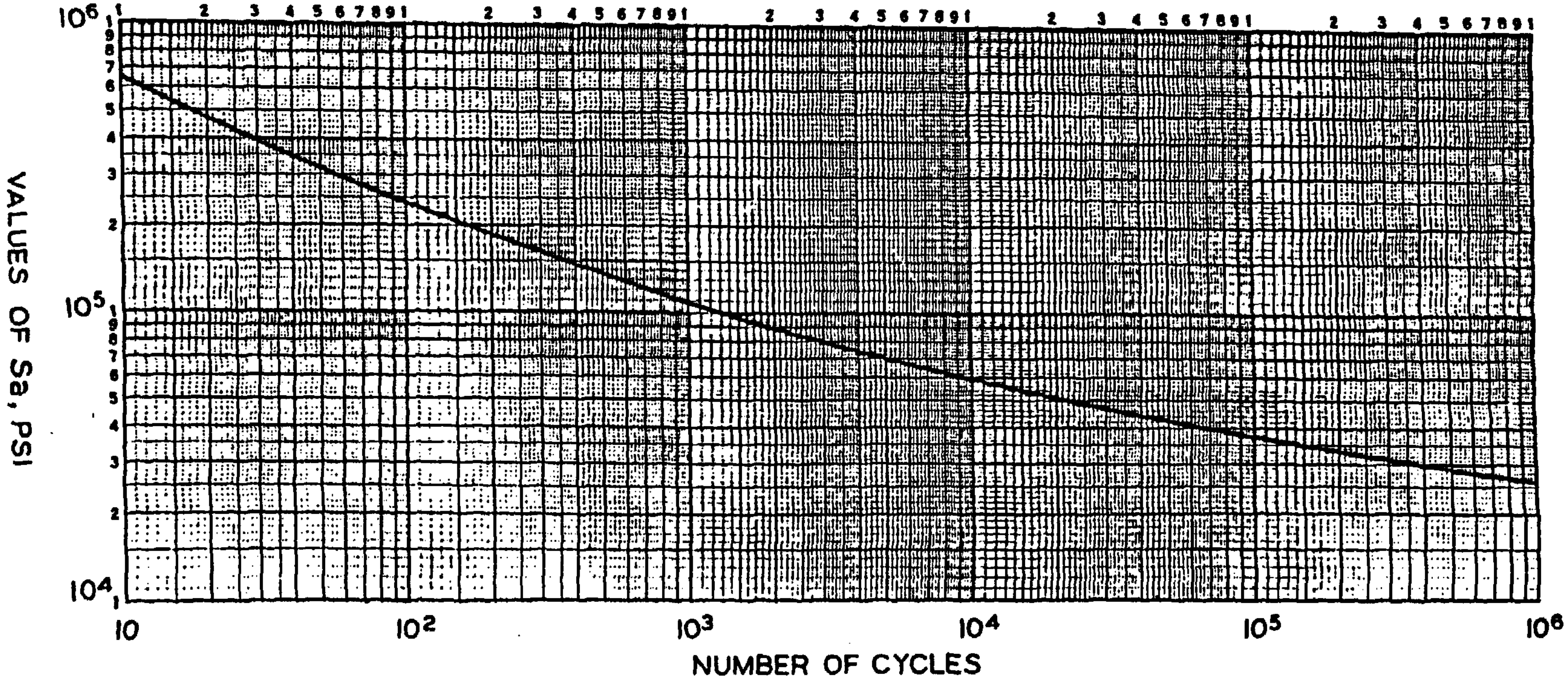


FIG. N-415(B) ALLOWABLE AMPLITUDE OF ALTERNATING STRESS, INTENSITY, S_a, FOR 18-8 STAINLESS STEELS AND (NICKEL-IRON CHROMIUM ALLOY) NICKEL-CHROME-IRON ALLOY (FOR METAL TEMPERATURES NOT EXCEEDING 800 F).