4.0 TIME-LIMITED AGING ANALYSES

Two areas of plant technical assessment are required to support an application for a renewed operating license. The first area of technical review is the Integrated Plant Assessment, which is described in Sections 2 and 3 of this License Renewal Application. The second area of technical review required is the identification and evaluation of plant-specific time-limited aging analyses and exemptions. The identifications and evaluations included in this section meet the requirements contained in 10 CFR 54.21(c) and provide the information necessary for the NRC to make the finding contained in 10 CFR 54.29(a)(2).

4.1 Identification of Time-Limited Aging Analyses

Title 10 of the Code of Federal Regulations, Part 54 (10 CFR 54) sets forth the requirements for License Renewal of Operating Nuclear Power Plants. Part 54.21(c)(1) of Title 10 requires a listing and an evaluation of Time-Limited Aging Analyses (TLAAs). Part 54.21(c)(2) requires a listing and evaluation of active plant-specific exemptions granted under 10 CFR 50.12 that are based on TLAAs as defined in Part 54.3(a).

4.1.1 Identification Process of Time-Limited Aging Analyses

This section documents the identification and disposition of Time-Limited Aging Analyses (TLAAs), including TLAA related exemptions granted in accordance with 10 CFR 50.12, which are applicable to Ginna Station for the period of extended operation.

Time-limited aging analyses are defined in 10 CFR 54.3 as those licensee calculations and analyses that meet the following criteria:

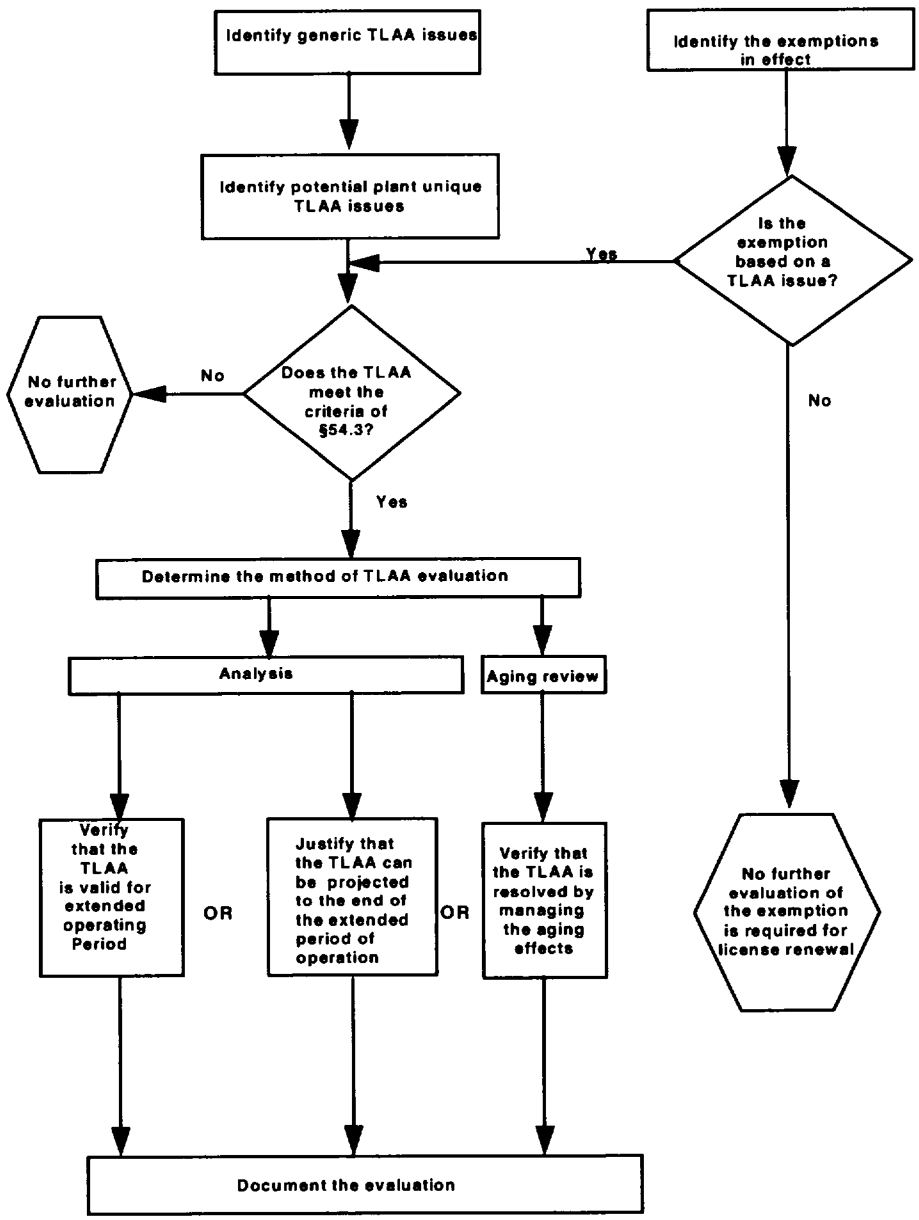
- Involve systems, structures, and components within the scope of license renewal, as delineated in 10 CFR 54.4(a);
- Consider the effects of aging;
- Involve time-limited assumptions defined by the current operating term, for example, 40 years;
- Were determined by the licensee to be relevant in making a safety determination;
- Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in10 CFR 54.4(b); and
- Are contained or incorporated by reference in the current licensing basis.

Some evaluations originally included in the Ginna Station Licensing Basis as TLAAs were re-evaluated within the current licensing period and were subsequently shown not to be time-limited for the 60 year operating period. Those evaluations have been included in this section to provide reviewers with a clear understanding of how these potential TLAAs were resolved. Affected TLAAs include Reactor Vessel Nozzle-to-Vessel Weld Defect (Section 4.3.5), Pressurizer Fracture Mechanics Analysis (Section 4.3.6) and RCP Flywheel (Section 4.7.6).

4.1.2 TLAA Methodology

This section discusses the methodology used to complete this process. Ginna Station engineering procedure EP-3-S-0717 provides the instructions for performing this identification and evaluation. The methodology provided in the procedure is consistent with NEI 95-10 (Reference 4) and NUREG-1800 (Reference 3). The overall methodology is provided in (Figure 4.1-1).

Figure 4.1-1 TLAA Methodology



Potential TLAAs are identified in two ways:

- Reviewing lists of previously identified TLAAs and choosing those generically applicable to Ginna Station for further evaluation. Primary sources of information include the following:
  - NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (Reference 3)
  - NEI 95-10 [Revision 3], "Industry Guideline for Implementing the Requirements of 10 CFR 54 -the License Renewal Rule" (Reference 4)
  - Previous License Renewal Applications (up to October 2001)
- Searching the Ginna Station CLB for calculations/analyses which meet the definition of a TLAA. The primary sources of information include the following:
  - UFSAR
  - Technical Specifications
  - NRC Correspondence (includes SERs)
  - Gilbert and Westinghouse Correspondence
  - Configuration Management Information System (CMIS)
  - Commitment Action Tracking System (historical database)

Key Word Searches

The Ginna Station CLB documents are available in electronic format or have been electronically indexed and were keyword searched. To ensure a complete review was been performed, Engineering subject matter experts reviewed the results to provide additional assurance that all potential TLAAs were identified.

Life	Lifetime
EFPY	License
Erosion Allowance	Corrosion Allowance
Forty Years	40 Years
Cycle	Fatigue Analysis (within 10 words of) year

Documenting the search process:

For each potential TLAA, a form is initiated. This includes the assignment of a tracking number, the document under review, the document date (if applicable), the subject, and some specific information which indicates why the document was "flagged" for review.

4.1.3 Identification and Evaluation of Active Plant-specific Exemptions

All exemptions are contained within the Ginna Station Operating License. All active NRC exemptions granted under 10 CFR 50.12 were identified, documented as potential TLAAs and evaluated. No TLAA-related exemptions were found.

4.1.4 Screening of Potential Time-Limited Aging Analyses

The process of screening was performed to make a final determination of which issues/documents would be evaluated per §54.21(c)(1). This was done by applying the six criteria delineated in §54.3. The guidance used for applying the six criteria is provided below.

1. Involve systems, structures, and components within the scope of license renewal as delineated in §54.4(a). The system, structure, and component scoping step of the IPA should be performed prior to or concurrent with the TLAA identification. Alternatively, the LRE may use engineering judgement to consider a SSC within the scope of the rule using a bounding approach without a specific evaluation against §54.4(a).

2. Consider the effects of aging. The effects of aging include but are not limited to: loss of material, loss of toughness, loss of prestress, settlement, cracking, and loss of dielectric properties.

3. Involve time-limited assumptions defined by the current operating term, for example 40 years. The defined operating term should be explicit in the analysis. Simply asserting that a component is designed for a service life or plant life is not sufficient. A calculation or analysis that explicitly includes a time limit must support the assertion.

4. Were determined relevant in making a safety determination. Relevancy is a determination that is made based on a review of the information available. A calculation or analysis is relevant if it can be shown to have direct bearing on the action taken as a result of the analysis performed. Analyses are also relevant if they provide the basis for a safety determination, and in the absence of the analyses, a different conclusion may have resulted.

5. Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, or component to perform its intended functions as delineated in §54.4(b). As stated in the first criterion, the intended functions must be identified prior to or concurrent with the TLAA identification. Analyses that do not affect the intended functions of the system, structure, or components are not TLAAs.

6. Are contained or incorporated by reference in the CLB. Plant specific documents contained or incorporated by reference in the CLB include the UFSAR, SERs, Technical Specifications, the fire protection plan/hazards analyses, correspondence to and from the NRC, QA plan, or topical reports included as reference to the UFSAR. Calculations and analyses that are not in the CLB or not incorporated by reference are not TLAAs.

4.1.5 Evaluation Process of Time-Limited Aging Analyses

Once a TLAA was identified, an evaluation was performed, as required by 10 CFR 54.21(c)(1), to demonstrate that at least one of the following criteria is applicable:

i. The analyses remain valid for the period of extended operation.

ii. The analyses have been projected to the end of the period of extended operation.

iii. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

The results of these evaluations are provided in Table 4.1-1 and are discussed in Sections 4.2 through 4.7.

Table 4.1-1 Time-Limited Aging Analysis Categories

TLAA Category	Analysis Description	Disposition	Report Section
Reactor Vessel Neutron Embittlement	Upper-Shelf Energy	(ii)	Section 4.2.1
	Pressurized Thermal Shock	(ii)	Section 4.2.2
	Pressure-Temperature (P-T) Limits	(ii)	Section 4.2.2
Metal Fatigue	ASME Boiler and Pressure Vessel Code Section III, Class 1	(i)	Section 4.3.1
	Reactor Vessel Underclad Cracking	(ii)	Section 4.3.3
	ANSI B31.1	(i)	Section 4.3.2
	Accumulator Check Valve	(i)	Section 4.3.4
	Reactor Vessel Nozzle-to-Shell Weld Defect	(i)	Section 4.3.5
	Pressurizer Fracture Mechanics Analysis	(i)	Section 4.3.6
	Environmentally Assisted Fatigue	Not a TLAA	Section 4.3.7
Environmental Qualification of Electrical Equipment	EEQ Evaluations	(i), (ii), or (iii) depending on the TLAA related components	Section 4.4
Concrete Containment Tendon Prestress	Concrete Containment Tendon Prestress	(iii)	Section 4.5
Containment Liner Plate and Penetration Fatigue	Containment Liner Plate and Penetration Fatigue	(ii)	Section 4.6

Table 4.1-1 Time-Limited Aging Analysis Categories

Other Plant Specific TLAAs	Containment Liner Stress	(iii)	Section 4.7.1
	Containment Tendon Fatigue	(i)	Section 4.7.2
	Containment Liner Anchorage Fatigue	(i)	Section 4.7.3
	Containment Tendon Bellows Fatigue	(i)	Section 4.7.4
	Crane Load Cycle Limit	(i)	Section 4.7.5
	Reactor Coolant Pump Flywheel	(iii)	Section 4.7.6
	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	(ii)	Section 4.7.7

4.2 Reactor Vessel Neutron Embrittlement

The reactor vessel is subjected to neutron irradiation from the core. This irradiation results in the embrittlement of the reactor vessel materials. A neutron fluence calculation has been performed as part of the RCS pressure-temperature operating limits analysis and subsequently been used as a basis for fluence values used in other Reactor Vessel Neutron Embrittlement Analyses. The methodology used to perform neutron fluence calculations in consistent with Regulatory Guideline 1.190. Analyses have been performed that address the following:

- Upper shelf energy
- Pressurized thermal shock
- RCS pressure-temperature operating limits

4.2.1 Upper Shelf Energy

Introduction

The Charpy upper shelf energy is associated with the determination of acceptable Reactor Vessel toughness during the license renewal period. 10 CFR Part 50 Appendix G paragraph IV.A.1 requires that the reactor vessel beltline materials must have a Charpy upper shelf energy of no less than 68 J (50 ft-lb) throughout the life of the reactor vessel, unless otherwise approved by the NRC. Surveillance capsules are attached to the inside of the RPV at locations designed to provide a higher irradiation rate, thus providing an irradiation "lead" factor that allows for prediction of future vessel irradiation damage.

In the event that the 50 ft-lb requirement cannot be satisfied as stated in 10 CFR 50 Appendix G, or by alternative procedures acceptable to the NRC, the reactor vessel may continue to operate provided requirement 1 of Appendix G is satisfied. This requirement states that an analysis must conservatively demonstrate, making appropriate allowances for uncertainties, the existence of equivalent margins of safety for continued operation.

Conclusion

The upper shelf energy for the reactor vessel beltline weld material at the end of the extended period of operation is expected to decrease to less than 50 ft-lbs based on predictions using RG 1.99. A low upper-shelf fracture mechanics analysis has been performed (Reference 22) to evaluate the weld material for ASME Levels A, B, C, and D Service Loadings, based on the acceptance criteria of the ASME Code, Section XI, Appendix K.This analysis follows the same approach previously approved by the NRC in BAW-2192PA and BAW-2178PA (Reference 23 and Reference 24). The analysis demonstrates that the limiting reactor vessel beltline weld satisfies the ASME Code

Requirements of Appendix K for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 52 effective full power years of plant operation. Therefore the analysis associated with upper-shelf energy has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.2.2 Pressurized Thermal Shock

Introduction

The PTS rule, 10 CFR 50.61 established screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature RT_{PTS}. The screening criteria are 270°F for plates and axial welds, and 300°F for circumferential welds. The RT_{PTS} is defined as:

$RT_{PTS} = I + \Delta RT_{NDT} + M$

Where: I = Initial reference temperature

ΔRT_{NDT} = Mean value of adjustment in reference temperature

M = Margin

The initial reference temperature is the measured unirradiated value as defined in the ASME Code, Paragraph NB-2331. If measured values are unavailable for the heat of the material of interest, generic values may be used. The generic values are based on the data for materials of all heats that were made by the same vendor using similar processes. The generic values of initial reference temperature for welds are defined in the PTS rule and used in this analysis for conservatism.

The ΔRT_{NDT} depends upon the amount of neutron irradiation and the amounts of copper and nickel in the material. It is calculated as the product of a fluence factor and a chemistry factor. The fluence factor is calculated from the best-estimate neutron fluence at the interface of cladding, weld, and metal on the inside surface of the vessel at a location where the material receives the highest fluence at the end of the period of evaluation. The fluence value used in this analysis is based on the results of calculations performed following the guidance in Regulatory Guide 1.190. The chemistry factor may be determined using credible surveillance data or from the chemistry factor tables in the PTS rule; these tables are used in this analysis for conservatism.

The margin term is intended to account for variability in initial reference temperature and the adjustment in reference temperature caused by irradiation. The value of the margin term is dependent of whether the initial reference temperature was a measured or generic value and whether the adjustment in reference temperature was determined from credible surveillance data or from the chemistry factor tables in the PTS rule. For the purpose of this analysis, the margin term will be based on a generic value for welds fabricated using Linde 80 flux as documented in B&W Owners Group report BAW-1803, Rev. 1.

Conclusion

The results of the revised PTS analysis for the limiting material have been reviewed for compliance with 10 CFR 50.61. The methodology used in PTS analysis is based on the projected neutron fluence at the end of the period of extended operation (52 EFPY) and does not rely on plant-specific surveillance data to calculate RT_{PTS}. Although plant-specific surveillance data could have been used, generic data calculated in accordance with Regulatory Guide 1.99, Rev. 2, Position 1.1 proved to be more conservative.

Table 4.2-1 Values of RT_{PTS} at 52 EFPY - Ginna RPV Beltline Materials

Material	Heat Number	Inner Surface Fluence E19 n/cm²	Initial RT_{NDT} °F	Margin °F	Chemistry Factor °F	Inside Surface Fluence Factor	ΔRT_{NDT} °F	RT_{PTS} °F
Intermediate Shell	125S255VA1	4.85	20	34[1]	44[1]	1.396	61.4	115.4
Lower Shell	125P666VA1	4.85	40	34[1]	31[1]	1.396	43.3	117.3
Circumferential Weld	61782/ SA-847	4.85	-4.8	56[1]	170.4[1]	1.396	237.9	289.1

[1]Regulatory Guide 1.99, Rev. 2, Position 1.1

The RT_{PTS} values for the intermediate and lower shell forgings remain below the NRC screening criterion of 270°F and the RT_{PTS} value for the beltline circumferential weld (SA-847) remains below the NRC screening criterion of 300°F at 52 EFPY. The analysis associated with PTS has been projected to the end of the period of extended operation and is consistent with 10 CFR 54.21(c)(1)(ii)

4.2.2 Pressure-Temperature (P-T) Limits

Introduction

10 CFR Part 50 Appendix G requires that the reactor pressure vessel (RPV) be maintained within established pressure-temperature (P-T) limits including during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure (given the required minimum temperature) is reduced.

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT} and adding a margin.

Conclusion

The reactor vessel neutron fluence values corresponding to the end of the period of extended operation and the reactor vessel beltline material properties have been calculated consistent with Regulatory Guide 1.190. The revised fluence values have been used to determine the limiting value of RT_{NDT} using the methods of Regulatory Guide 1.99. The limiting value of RT_{NDT} was used to calculate reactor coolant system (RCS) pressure-temperature (P-T) operating limits that are valid through the end of the period of extended operation (Reference 25). P-T Curves were developed using ASME Code Case N-641, which allows for the use of the KIC methodology (ASME Code Case N-640) and the relaxed "Circ Flaw" methodology (ASME Code Case N-588). Consistent with NUREG-1800 section 4.2.2.1.3.3, it is not necessary to implement P-T limits to carry the reactor vessel through 60 years at the time of application. The updated limits must be contained in a pressure-temperature limit report (PTLR) or in the Technical Specification (TS) prior to the period of extended operation.

The analyses associated with reactor vessel pressure-temperature limits will be available prior to entering the period of extended operation, consistent with 10 CFR 54.21(c)(1)(ii).

4.3 Metal Fatigue

Although fatigue is not necessarily time-limited in the same manner as other TLAAs (since design limits are based on cycles and not an explicit time period), it has been identified by the NRC and previous license renewal applicants as a TLAA. There are two aspects to fatigue life evaluation. The first is fatigue design, which is based on transient cycles and is a TLAA and part of the plant CLB. The second is the consideration of reactor water environmental effects on fatigue life, as described in GSI-190, and is not part of the CLB. The TLAAs on fatigue design have been resolved by projecting that the original transient design cycles remain valid for the 60-year operating period. Reactor water environmental effects on fatigue life are evaluated using the most recent data from laboratory simulation of the reactor coolant environment. Environmentally-assisted fatigue effects are addressed by the Fatigue Monitoring Program.

4.3.1 ASME Boiler and Pressure Vessel Code, Section III, Class 1

Introduction

The reactor vessel, pressurizer, steam generators, and reactor coolant pumps have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class 1. The reactor vessel internals were designed in accordance with Westinghouse criteria which were later incorporated into the ASME Boiler and Pressure Vessel Code. The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that initiation of fatigue cracks is precluded.

Fatigue usage factors for critical locations in the NSSS components were determined using design cycles that were specified in the plant design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were subsequently utilized in the design stress reports for various NSSS components satisfying ASME fatigue usage design requirements, and became part of the plant Technical Specifications.

Experience has shown that actual plant operation is often very conservatively represented by these design cycles. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the Class 1 component fatigue analyses remain valid for the period of extended operation, the design cycles applicable to the Class 1 components were assembled. The actual frequency of occurrence for the design basis cycles was determined and compared to the design cycle set. The severity of the actual plant transients was compared to the severity of the design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis the design-cycle profiles envelope actual plant operation. In addition, a review of the applicable administrative and operating procedures was performed to verify the effectiveness of the current design cycle counting program.

Conclusion

This review concluded that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation. The analyses associated with verifying the structural integrity of the reactor vessel, reactor vessel internals, pressurizers, steam generators, and reactor coolant pumps have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

Prior to the expiration of the current operating license, a Fatigue Monitoring Program will be implemented as a confirmatory program.

4.3.2 ANSI B31.1 Piping

Introduction

The Reactor Coolant System primary loop piping and balance-of-plant piping were originally designed to the requirements of USAS B31.1, Power Piping Code. The pressurizer surge line was reanalyzed in 1991 and is treated separately in Section 4.3.7.

Design requirements in ANSI B31.1 assume a stress range reduction factor to provide conservatism in the piping design to account for fatigue due to thermal cyclic operation. This reduction factor is 1.0 provided the number of anticipated cycles is limited to 7000 equivalent full temperature cycles. This represents a condition where a piping system would have to be cycled approximately once every 3 days over the extended plant life of 60 years. Considering this limit, a review of the ANSI B31.1 piping within the scope of license renewal was performed in order to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal (except for the NSSS Sampling System) are only occasionally subject to cyclic operation. Typically these systems are subject to continuous steady-state operation and operating temperatures only vary during plant heatup and cooldown, during plant transients, or during periodic testing.

Conclusion

The results of the evaluation for ANSI B31.1 piping systems (except for the NSSS Sampling System) demonstrate that the number of assumed thermal cycles will not be exceeded in 60 years of plant operation. The analyses associated with ANSI B31.1 piping fatigue have been evaluated and determined to remain valid for the period of extended operation, in accordance with CFR 54.21(c)(1)(i).

For the NSSS Sampling System, an engineering analysis will be performed prior to the end of the current license period to verify that the allowable piping stresses (accounting for a stress range reduction factor less than 1.0) will not be exceeded during the period of extended operation. If the results of this analysis are not acceptable, an approach will be developed which will include one or more of the following options:

- Further refinement of the fatigue analysis;
- Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC); or
- Replacement of the affected components.

4.3.3 Reactor Vessel Underclad Cracking

Introduction

Underclad cracking is associated with reactor pressure vessel. The industry has reported cracking in the low-alloy base metal heat-affected zone (HAZ) beneath the austenitic stainless steel weld overlay that is deposited to protect the ferritic material from corrosion.

A detailed analysis of underclad cracks was provided by Westinghouse in topical report WCAP-7733 (Reference 1). This report justified the continued operation of Westinghouse plants for 32 effective full power years with underclad cracks in the vessel.

A re-evaluation of the underclad cracking issue for 60 years of plant operation was performed in WCAP-15338 (Reference 2) and concluded that "underclad cracks in a reactor vessel are of no concern relative to the structural integrity of the vessel for continued plant operation, even through 60 years of operation." The NRC reviewed WCAP-15338 and included two applicant action items to verify that a plant is bounded by the report evaluation and that the TLAA be described in the plant FSAR supplement.

Conclusion

WCAP-15338 is bounding for all Westinghouse plants and the Underclad Cracking TLAA is described in the UFSAR supplement. Therefore, the two applicant action items have been resolved.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

4.3.4 Accumulator Check Valves

Introduction

Anamet Laboratories report 172.1 describes analyses performed on the 10-C48Z Self-Actuating Swing Check Valves manufactured by the Darling Valve Co. and used in conjunction with the Ginna Station accumulators. Fatigue of components is recognized as time-dependent and therefore the analysis was reviewed for fatigue related to these valves. Fatigue failure is based upon the criteria of the cumulative usage factor (CUF). The Anamet report concludes that the maximum CUF is 0.74. The analysis is based on load condition occurrence limits provided by Westinghouse Electric Corp.

Conclusion

The 2000 Transient Monitoring Report was reviewed to confirm transient limits and total transient counts to date. The load condition occurrences used in the Anamet report bound the transient limits monitored by plant procedures. In accordance with 10 CFR 54.21(c)(1)(i), the existing CUF calculations remain valid because the number of assumed transients would not be exceeded during the period of extended operation.

4.3.5 Reactor Vessel Nozzle-to-Vessel Weld Defect

Introduction

In 1979, during the first-interval ISI of the reactor vessel, a flaw indication was discovered by UT examination in a primary inlet nozzle-to-vessel weld (Nozzle N2B). The size of this indication (5.27" long X .93" deep) was determined to be in excess of the size permitted by the acceptance criteria for the examination method in ASME Section XI, 1974 Edition. As a result, the flaw indication was evaluated by Teledyne Engineering Services in accordance with the Section XI (Appendix G) requirements for acceptance by evaluation and found to be acceptable. A review of original construction radiographs confirmed the presence of slag at the same location as the indication.

The same flaw indication was again recorded in Nozzle N2B by UT examination during the second-interval reactor vessel ISI performed by Southwest Research Institute (SwRI) in 1989. The size was again found to exceed the acceptance criteria in Section XI. However, using a 15° focused beam search unit, the indication was resolved into two separate indications which met the criteria for acceptance by examination in ASME Section XI, 1974 edition with Summer 1975 Addenda. A fracture mechanics analysis performed by Structural Integrity Associates also confirmed that the indication was acceptable by evaluation according to the requirements of ASME Section XI (Appendix G).

Conclusion

A safety evaluation performed by the NRC (Reference 26) concluded that the flaw indication was probably a volumetric reflector resulting from the fabrication process that had remained unchanged since construction and that augmented ISI required by the ASME Code was not warranted. Therefore, no further evaluation of this defect is required.

4.3.6 Pressurizer Fracture Mechanics Analysis

Introduction

During the preservice UT examination of the pressurizer, a "defect-like" indication was reported in the lower shell-to-head circumferential weld (C-3). The indication was reported as a linear reflector approximately 11 1/2" long X 1/2" width embedded partially in the circumferential weldment and the base metal of the pressurizer shell. Based on a fracture mechanics analysis performed by Westinghouse, it was concluded that the "defect" would not cause failure of the pressurizer during the design life (40 years) of the component. The analysis was based on several conservative assumptions, including transposing the defect from the embedded position to the internal surface of the pressurizer wall.

This indication was subsequently examined by UT in 1971, 1972, 1974, 1980, 1991, and 2002 during ASME Section XI inservice inspections. The examinations in 1974, 1980, and 1991 characterized the indication as consisting of several intermittent, low-amplitude indications located in the center 1/3 of the weld thickness. These indications were evaluated and found to meet the acceptance criteria by examination of ASME Code, Section XI. The most recent inspection was performed using both automated and manual UT examinations. Intermittent, low-amplitude indications were recorded in the center 1/3 of the weld thickness. These indications were also evaluated and found to meet the acceptance criteria by examination in ASME Code, Section XI, 1995 Edition (1996 Addenda).

Conclusion

Since it has been demonstrated that the initial indication is actually a number of small, discrete indications which meet the ASME Code, Section XI acceptance criteria by examination, the fracture mechanics analysis is no longer applicable or relevant.

4.3.7 Environmentally Assisted Fatigue

Generic Safety Issue (GSI) 190 was initiated by the NRC staff because of concerns about the potential effects of reactor water environments on reactor coolant system component fatigue life during the period of extended operation. The NRC closed GSI-190 in December 1999 and concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required. However, as part of the closure of GSI-190, NRC concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs (Reference 5). See (Figure 4.3-1) for a graphic depiction of the approach used for addressing the fatigue TLAAs and the Environmentally-Assisted Fatigue evaluation

An analysis must satisfy all six criteria defined in 10 CFR 54.3 to qualify as a TLAA. Failure to satisfy any one of these criteria eliminates the analysis from further consideration as a TLAA. Fatigue design analysis for Ginna Station has been determined to be a TLAA, even though the design limits are based on cycles rather than an explicit time period. However, reactor water environmental effects, as described in GSI-190, are not included in the Ginna Station current licensing basis (CLB), such that the criterion specified in 10 CFR 54.3(a)(6) is not satisfied.

Nevertheless, environmental effects on Class 1 component fatigue have been evaluated separately for Ginna Station to determine if any additional actions are required for the period of extended operation.

The Fatigue Monitoring Program addresses the effects of the reactor coolant environment on component fatigue life by assessing the impact of reactor coolant environment on a selected set of critical component locations that includes those components identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components" (Reference 6) for the older vintage Westinghouse plant. This evaluation can take one of two approaches; (1) demonstration that the fatigue analysis of design transients, when compared to an evaluation based on actual transients, will bound any environmental effects during the extended operating period, or (2) assessment of actual expected fatigue usage factor, specifically including the influence of environmental effects.

The second approach evaluates the selected plant component location fatigue usage utilizing the environmental life correction factor formulae contained in NUREG/CR-6583

"Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels" (Reference 7) for carbon and low-alloy steels and NUREG/CR-5704 "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels" (Reference 8) for austenitic stainless steels. These formulae are then used in an F_{en} approach, originally developed by EPRI, in which an environmental fatigue multiplier (F_{en}) is computed when certain conditions of dissolved oxygen, temperature, strain rate, strain range, sulfur content and flow rate are satisfied. The Fatigue Monitoring Program uses the second approach for the selected set of component locations

Approach for Addressing the Fatigue TLAAs and the Environmentally-Assisted Fatigue Evaluation

The approach for addressing fatigue TLAAs and reactor water environmental effects at Ginna Station accomplishes two objectives, as illustrated in (Figure 4.3-1). First, the TLAAs on fatigue design have been resolved by projecting that the original transient design cycles remain valid for the 60-year operating period. Confirmation by the Fatigue Monitoring Program will ensure that these transient design cycles are not exceeded. Second, reactor water environmental effects on fatigue life are examined using the most recent data from laboratory simulation of the reactor coolant environment. These two aspects of fatigue design are kept separate, because fatigue design for Ginna is part of the plant CLB and a TLAA, while the consideration of reactor water environmental effects on fatigue life, as described in GSI-190, is not considered part of the Ginna CLB.

It is important to note that there are three areas of margin included in the Ginna Fatigue Monitoring Program that are worthy of consideration. These areas include margins resulting from actual cycle experience, cycle severity, and moderate environmental effects.

Margin Due to Actual Cycles: It has been concluded that the original 40-year design cycle set for Class 1 components is valid for the 60-year extended operating period. Conservative projections conclude that the design cycle limits will not be exceeded. Additional margin is available in the current Class 1 component fatigue analyses since the cumulative fatigue usage factors for all Class 1 components remain below the acceptance criteria of 1.0.

Margin Due to Transient Severity: Much of the conservatism in the fatigue analysis methodology is due to design cycle definitions. It has been concluded that the severity of the original Ginna design cycles bound actual plant operation. Additional industry fatigue studies conclude that the fatigue impact of conservative design basis cycle definitions by themselves overwhelms the contributing impact of reactor water environmental effects.

Margin Due to Moderate Environmental Effects: A portion of the safety factors applied to the ASME Code Section III fatigue design curves includes moderate environmental effects. While there is debate over the amount of margin this represents, it is noteworthy to recognize this safety factor in this qualitative discussion of margin.

Considering the three margins above, the Ginna Fatigue Monitoring Program is conservative from an overall perspective. Nevertheless, specific assessments of potential environmental effects have been addressed. Idaho National Engineering Laboratories (INEL) evaluated in NUREG/CR-6260 (Reference 6) fatigue-sensitive component locations at plants designed by all four U. S. nuclear steam supply system (NSSS) vendors, as a part of the industry effort to address environmental effects for operating nuclear power plants during the current 40-year licensing term. The pressurized water reactor (PWR) calculations included in NUREG/CR-6260, especially the "Older Vintage Westinghouse Plant", bound Ginna Station with respect to the design codes used. Additionally, the evaluated design cycles considered in the evaluation bound the Ginna Station design.

The fatigue-sensitive component locations chosen in NUREG/CR-6260 for the older vintage Westinghouse plant were:

- Reactor vessel shell and lower head (lower shell at the core support pads)
- Reactor vessel inlet and outlet nozzles
- Pressurizer surge line (including hot leg and pressurizer nozzles)
- Reactor coolant piping charging system nozzle
- Reactor coolant piping safety injection nozzle
- Residual Heat Removal system Class 1 piping

NUREG/CR-6260 calculated fatigue usage factors for these locations utilizing the interim fatigue curves provided in NUREG/CR-5999 (Reference 10). However, the data included in more recent industry studies (Reference 7 and Reference 8) need to be considered in the evaluations of environmental effects.

Environmental fatigue calculations have been performed for Ginna Station for those component locations included in NUREG/CR-6260 using the appropriate methods contained in NUREG/CR-6583 for carbon/low alloy steel material, or NUREG/CR-5704 for stainless steel material. Based on these results, all component locations were determined to be acceptable for the period of extended operation, with the exception of the pressurizer surge line.

4.3.7.1 Reactor Vessel Locations

Appropriate F_{en} factors were computed for each individual load pair in the governing fatigue calculation so that an overall multiplier on the fatigue usage factor for environmental effects was determined for each component. Application of these factors to the design fatigue usage resulted in acceptable values for the period of extended operation. The locations analyzed are the RPV inlet nozzles, RPV outlet nozzles and the RPV Shell at the core support pads.

4.3.7.2 Surge Line Locations

A structural evaluation of the Ginna surge line considering the effects of thermal stratification was performed by Westinghouse in 1991 (Reference 12). WCAP-12928 describes the stress and cumulative usage factor analysis performed for the surge line in accordance with NRC Bulletin 88-11. The highest CUF was calculated at the surge line nozzle connection to the RCS hot leg.

The F_{en} approach was not able to produce acceptable results for the period of extended operation due to the significant thermal cycling duty and high environmental fatigue multipliers. The environmentally-adjusted fatigue usage value for the limiting pressurizer surge line location is calculated to exceed 1.0 before the end of the period of extended operation.

An aging management program will be used to address environmentally-assisted fatigue for the Ginna pressurizer surge line during the period of extended operation. Prior to the end of the current license period, critical weld locations (i.e. pressurizer surge line nozzle weld, surge line reducer-to-pipe weld below the pressurizer, and the RCS hot leg surge line nozzle weld) on the surge line will be inspected in accordance with the appropriate requirements of IWB under the ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection Program. Prior to the period of extended operation, the results of these inspections and research planned by the EPRI-sponsored Materials Reliability Program (MRP) will be used to determine the appropriate approach for addressing environmentally-assisted fatigue of the surge line. The approach developed will include one or more of the following options:

- Further refinement of the fatigue analysis to lower the CUF to below 1.0;
- Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC);
- Repair of the affected location(s); or
- Replacement of the affected location(s).

Should RG&E select the inspection option to manage environmentally-assisted fatigue for the surge line during the period of extended operation, inspection details such as scope, qualification, method, and frequency will be provided to the NRC for review and approval prior to entering the period of extended operation. This position is consistent with previous applicants' positions.

Various pressurizer surge line welds at locations of high fatigue usage have been examined at Ginna Station in the past. No reportable indications were found by these examinations. The absence of any reportable indications on the surge line to-date supports the use of the aging management program outlined above.

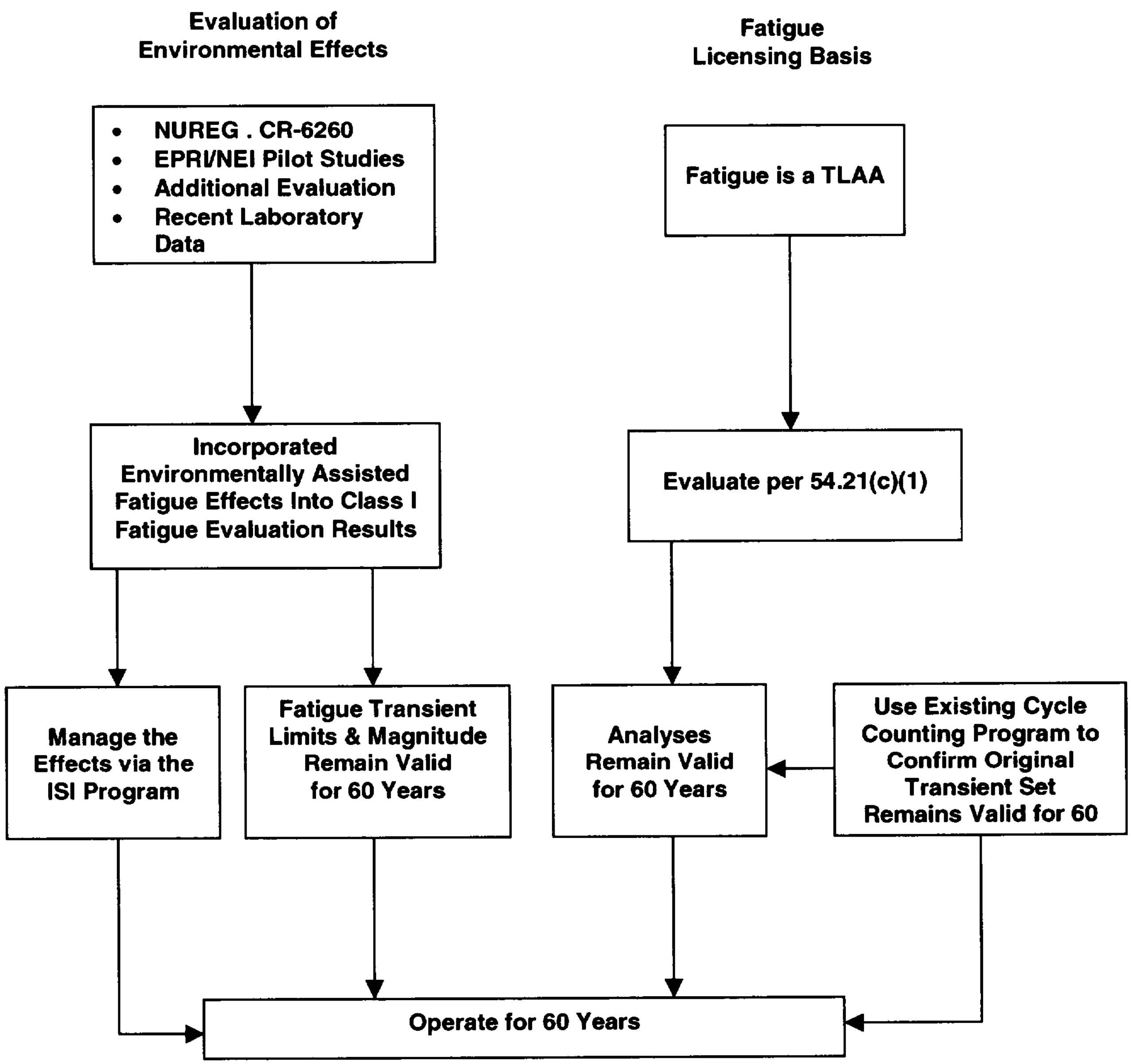
4.3.7.3 USAS B31.1 Locations

As with the older vintage Westinghouse plants evaluated in NUREG/CR-6260, detailed fatigue usage calculations do not exist for the Ginna Station RHR tee, charging nozzle and safety injection (accumulator) nozzle, because the design basis for the Ginna Station piping is USAS B31.1, which does not require specific fatigue analysis. In response to this circumstance, the authors of NUREG/CR-6260 developed a detailed ASME Class 1 fatigue analysis for each of these three components based on typical inputs These analyses were used to develop Ginna-specific environmental fatigue calculations. The design transients and cycle counts in NUREG/CR-6260 bound the design transients and cycle counts for Ginna Station for these three component locations. The design inputs for Ginna Station (e.g. material, geometry) were compared to those summarized in NUREG/CR-6260 for these three components. The Ginna Station charging nozzle and safety injection nozzle were determined to be identical in terms of materials and geometry as those presented in NUREG/CR-6260. The Ginna Station RHR tee is identical in terms of material, but is larger than the RHR tee analyzed in NUREG/CR-6260. The NUREG/CR-6260 RHR tee was determined to bound the stress ranges for the larger Ginna Station RHR tee. Adjustments to the design basis fatigue usage (without environmental effects) were made based on the design input comparison. Plant-specific F_{en} factors were computed and were applied to the Ginna-specific design basis fatigue usage to yield Ginna-specific environmental fatigue values. This process resulted in acceptable values for the period of extended operation.

Conclusion for Environmentally Assisted Fatigue

The Fatigue Monitoring Program utilizes 10 CFR 54.21(c)(1)(i) for each of the plant components within the scope of the program. The Fatigue Monitoring Program is a confirmatory program that monitors loading cycles due to thermal and pressure transients for selected critical components. The program provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation. The effects of reactor coolant environment are considered through the evaluation of the seven component locations identified in NUREG/CR-6260 using the appropriate environmental fatigue factors. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 (Reference 7) for carbon or low-alloy steels and in NUREG/CR-5704 (Reference 8) for austenitic stainless steels.

Figure 4.3-1 TLAA & GSI-190 Environmentally Assisted Fatigue Evaluation Process



4.4 Environmental Qualification (EQ) of Electric Equipment

10 CFR 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants, requires that selected electrical equipment that is relied upon to remain functional during and following a design basis event be environmentally qualified to perform its intended function. Equipment within the scope of the EQ rule has been identified in accordance with 10 CFR 50.49 paragraph (d) and is listed in the Ginna Station EQ Master List. Only the equipment qualification packages which indicate a qualified life of greater than 40 years will be reviewed as a Time-Limited Aging Analysis (TLAA). Equipment qualification packages that indicate a qualified life of less than 40 years are not a TLAA as defined in 10 CFR 54.3 and therefore will not be discussed in the context of license renewal.

To establish reasonable assurance that the safety related electrical equipment will perform its safety function when exposed to postulated harsh environmental conditions, licensees are required to develop an environmental qualification program. The program must demonstrate that the safety related electrical equipment required to perform the various safety related functions, identified in 10 CFR 50.49, is qualified to perform as intended. The program must maintain the environmental qualification of the equipment for its installed life. Periodic replacement and/or refurbishment of equipment is performed in order to maintain the qualified life of the device. The qualified life of an equipment type is that period of time the equipment can be installed, under normal and abnormal plant operating conditions (thermal and radiation exposure), and can still perform its intended function following a postulated design basis event. The qualified life of an equipment type is determined using the ambient environmental conditions to which it is exposed for the predicted installation period as well as any internal heat rise and cyclic stresses.

Many of the EQ analyses may have been adequate under existing conditions and therefore could have been dispositioned per 10 CFR 54.21(c)(1)(i), however it was felt to be conservative to perform a confirmatory evaluation to verify that the assumptions in the existing analysis were adequate for the period of extended operation. Confirmatory analyses do not alter any conservatisms, use data reduction methods or use different analysis methodology.

Existing EQ analyses have radiation levels for accident conditions based on a power level of 1520 MWt. However the UFSAR was revised to account for instrument uncertainty and therefore provided environmental tables were revised for 102% power or 1550.4 MWt. All EQ packages have been reviewed to verify that the margin between the required radiation qualification and the actual radiation qualification is adequate to cover the increase.

Table 4.4-1 contains a list of re-analysis attributes and methodology descriptions used for EQ TLAAs. Unless otherwise specified, each TLAA reviewed in accordance with 10 CFR

54.21(c)(1)(i) or 10 CFR 54.21(c)(1)(ii) (and 10 CFR 54.21(c)(1)(iii) in some cases) have been or will be performed using the methodology described in this table.

Table 4.4-1 Environmental Qualification Reanalysis Attributes

Reanalysis Attributes	Description
Analytical methods	The analytical models used in the reanalysis of an aging evaluation should be the same as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40 year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.
Data collection and reduction methods	Reducing excess conservatisms in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis. Temperature data used in an aging evaluation should be conservative and based on plant design temperatures or on actual plant temperature data. When used, plant temperature data can be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). A representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis should be justified. Similar methods of reducing excess conservatisms in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging.

Table 4.4-1 Environmental Qualification Reanalysis Attributes

Underlying assumptions	Environmental qualification component aging evaluations contain sufficient conservatisms to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected environmental qualification component is evaluated, and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.
Acceptance criteria and corrective actions	The reanalysis of an aging evaluation should extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component must be refurbished, replaced, or requalified prior to exceeding the current qualification. A reanalysis should be performed in a timely manner (such that sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful).

4.4.1 Solenoid Operated Valves

4.4.1.1 ASCO Solenoid Valve Model X-HAV210

Introduction

ASCO Solenoid Valve, Model X-HAV210 is used inside the Containment Building at Ginna Station in a single EQ solenoid operated valve (SOV) application. This SOV controls air supply to an air driven pump used to obtain post-accident samples at Sump A. Plant documents identify the SOV as installed in 1982. The critical components of the ASCO SOV that determine qualified life are the gaskets. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of elevated operating temperature during valve operation. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models, and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.2 Valcor Solenoid Operated Valve, Model V526-5440-2

Introduction

Valcor SOV, Model V526-5440-2 is used in the Intermediate Building at Ginna Station to control instrument air to main steam isolation valves (MSIVs). There are eight EQ SOVs, four providing redundant isolation and venting to each of two MSIVs. These SOVs are considered to be a part of the Main Steam system. They are located on the Mezzanine Level of the Intermediate Building in the steam header A and B areas. Plant documents identify these SOVs as installed in 1984 with one replacement in 1987. Valcor SOV, Model V526-5440-2, is a pilot assisted latching valve type with separate opening and closing coils. Valcor supplied rectifiers, Valcor Part No. S1140-8-1 associated with these valves are installed in terminal boxes separate from the valve housings, but are considered for purposes of qualification to be part of the valve due to the failure effects. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. The SOV coils are not continuously energized and are therefore not subject to significant self heating. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.3 Valcor Solenoid Operated Valve, Model V526-6130-2

Introduction

One Valcor SOV, Model V526-6130-2 is used in the Containment Building at Ginna Station to control containment Sump A sampling pump discharge flow. This direct lift, normally open two way SOV is considered to be a part of the Plant Sampling System (PSS). The SOV is located on the 235 foot level, basement Accumulator B area. Plant documents identify this SOV as installed in 1983. Valcor supplied rectifiers and diodes, Valcor Part No. S1140-8-1 associated with this valve are installed inside the valve electrical terminal housing.

Qualification for Valcor SOV, Model V526-6130-2 is performed in accordance with IEEE 323-1974 and includes the installed rectifier and diode configuration. The vendor provided qualification report states that all O-rings meet a 40 year qualified life. In anticipation of lack of control of service conditions for installed valves, Valcor recommends that elastomeric solenoid seals (O-rings) be replaced at shorter intervals. At Ginna Station SOV service conditions have been monitored and analyzed. Therefore, Valcor SOVs will be considered to be installed under controlled service conditions. However, due to thermal setting properties of EPR, O-rings will be replaced whenever the valve is disassembled after operation.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of elevated operating temperature during valve operation. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative.The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.1.4 Head Vent Solenoid Operated Valves

Introduction

Two Valcor model V526-6042-3 and two Valcor model V526-6042-17 solenoid operated valves (SOVs) are used to vent noncondensable gases from the reactor.These valves use the original electrical solenoid assembly, however two of the four valves have replacement valve bodies and therefore carry a different model number. Plant documents identify that these valves were installed in 1984 and qualified in accordance with IEEE 323-1974.

Conclusion

At this time, there are no plans to extend the qualification of the existing valves, and therefore the qualified portions of the valves will remain scheduled for replacement prior to the year 2022. A review of plant temperatures for the existing analyis provides reasonable assurance that the temperature used to determine qualified life remains conservative. In the event that reanalysis is performed, it will be completed consistent with the guidance provided in NUREG-1800 table 4.4-1. Therefore the environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21 (c)(1)(iii).

4.4.2 Motors

4.4.2.1 Westinghouse Containment Recirculation Fan Motor

Introduction

The containment recirculation fan motors are part of the containment recirculation fan cooling (CRFC) system. Five of these motors are rotated through the four CRFC units as necessitated by motor maintenance requirements. Four motors are original plant equipment, identified by Westinghouse shop order number 68F13557 and serial numbers 1S-69, 2S-68, 3S-68, and 4S-68. The fifth motor was initially installed in April of 1995 and is identified by Westinghouse shop Number 17570LN. All five are 300HP, 440VAC motors.

Qualification life for these motors is based on three critical components. These are winding insulation, bearings, and bearing lubrication. A motor bearing and its lubrication are treated as a single replaceable part, therefore winding insulation is the limiting component for evaluation of motor qualified life. These windings are insulated with Thermalastic Epoxy. The existing qualification is based on DOR guidelines.

Conclusions

The thermal and radiation analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of elevated operating temperature during motor operation. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to demonstrate qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.2.2 Limitorque Actuators, Outside Containment

Introduction

Nine EQ Limitorque Actuators are located in the Auxiliary Building basement. Four additional EQ units are located in the Auxiliary Building subbasement RHR pump pit. These actuators were put in service November 1969 and were originally qualified to DOR Guidelines. In the early 1990's, in response to NRC Generic Letter 89-10 all actuators were completely refurbished including replacement torque and limit switches qualified to IEEE 323-1974. All motors are Class B insulated with six units retaining their original motors. All internal wiring is replacement Anaconda by RG&E with separate qualification to IEEE 323-1974.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. Equipment operation is a negligible portion of service life and self heating due to temperature rise is not a significant contributor to age related degradation. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.2.3 Limitorque Actuators, Inside Containment

Introduction

Two EQ Limitorque Actuators are used on the RHR discharge valves located in the Containment Building basement. These actuators were placed in service in November of 1969 and were originally qualified to DOR guidelines. In the early 1990's, in response to NRC Generic Letter 89-10 all actuators were completely refurbished including replacement torque and limit switches qualified to IEEE 323-1974. All motors are replacement Class RH insulated and qualified to IEEE 323-1974. All internal wiring is Anaconda and Brand-Rex, replaced by RG&E with separate qualification to IEEE 323-1974.

Conclusion

The thermal, radiation, and cyclical wear analyses support a qualified life in excess of 60 years. Equipment operation is a negligible portion of service life and self heating due to temperature rise is not a significant contributor to age related degradation. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models, temperature, and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.2.4 Limitorque Actuators, PORV Block Valves

Introduction

Limitorque Model SMB-00-15 Motor Operated Valves (MOVs) with Reliance motors are used as block valves to prevent inadvertent depressurization of the primary system should the pressurizer power operated relief valves fail to close. These components are located in the pressurizer cubicle in the containment building. During the refueling outage of 1989, these MOVs were replaced because the seat rings were approching the maximum allowable limits for remachining The age degradable sub-components considered include the Fiberite torque switch, the type RH motor insulation, and the Viton shaft seal.

Conclusion

The previous thermal and radiation analyses support a qualified life of 40 years. The motors are only cycled for testing, and are not normally energized. Self-heating is not considered significant for these motor operated valves. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of installation and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.2.5 Westinghouse Safety Injection Pump 1A/1C Motor

Introduction

Westinghouse Model Lifeline A ABDP motors are installed in the Auxiliary Building and used to provide motive power to rotate the shaft of safety injection pumps 1A and 1C during the worst case design basis accident. The motors are original plant equipment and qualification is performed consistent with DOR guidelines.

Conclusion

Reanalysis of the motor qualified life is scheduled to be performed prior to the end of the current license period. Although the existing analysis indicates a qualified life of significantly greater than 60 years, the formal reanalysis will verify that the conditions and assumptions used in the original analysis remain valid for the period of extended operation. If the qualification cannot be extended by reananlysis, the motor will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.2.6 Westinghouse/Reliance Safety Injection Pump 1B Motor

Introduction

A Westinghouse Model ABDP motor was refurbished and qualified by the Reliance Electric Company. The motor is used to provide motive power to rotate the shaft of the safety injection pump during the worst case design basis accident. It was refurbished in 1994 and qualified to IEEE 323-1974.

Conclusion

The previous thermal and radiation analyses support a qualified life of 44 years. Thermal aging calculations consider the effects of elevated operating temperature during motor operations. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of installation and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.2.7 Westinghouse Containment Spray Pump Motors

Introduction

Westinghouse motor model Lifeline T TBDP is used to provide motive power to rotate the shaft of the containment spray pumps during the worst case design basis accident. The motors are installed in the Auxiliary Building basement and are normally de-energized. For qualified life calculations, it is conservatively assumed that these motors operate for 5 hours per year. This motor is original plant equipment qualified consistent with DOR guidelines.

Conclusion

Reanalysis of the motor qualified life will be performed prior to the end of the current license period. Although the existing analysis indicates a qualified life of significantly greater than 60 years, the formal reanalysis will verify that the conditions and assumptions used in the original analysis remain valid for the period of extended operation. If the qualification cannot be extended by reananlysis, the motor will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.2.8 Westinghouse RHR Pump 1A Motor

Introduction

Westinghouse motor model Lifeline T TBDP is used to provide motive power to rotate the shaft of the residual heat removal pump 1A during the worst case design basis accident. The motor is installed in the Auxiliary Building sub-basement. This motor is original plant equipment qualified consistent with DOR guidelines. The existing qualified life for this motor is 47 years. This life is calculated by conservatively assuming the operating time is 10% of total life. Since plant operating cycle time has been increased to 18 months, it is expected that this proportion will decrease.

Conclusion

Reanalysis of the motor qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the motor will be refurbished, replaced, or requalified prior to exceeding the current qualification period. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.2.9 Westinghouse RHR Pump 1B Motor

Introduction

Westinghouse motor model Lifeline T TBDP is used to provide motive power to rotate the shaft of the residual heat removal pump 1B during the worst case design basis accident. This component is installed in the Auxiliary Building sub-basement. The motor was refurbished by Westinghouse in 1989 and qualified to IEEE 323-1974.

Conclusion

The previous thermal and radiation analyses support a qualified life of 40 years. Thermal aging calculations consider the effects of elevated operating temperature during motor operations. Due to the date of installation and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.2.10 Westinghouse Hydrogen Recombiner Blower Motor

Introduction

Westinghouse motor model TBFC is used to provide motive power to rotate the shaft of the hydrogen recombiner blowers. This component is installed in the Containment Building and is normally de-energized. Although the original analysis calculates a possible operation time of 8 hours per month, this motor is not normally operated and therefore an assumption of 8 hours per year would be conservative. The motor is original plant equipment, qualified consistent with DOR Guidelines.

Conclusion

Reanalysis of the motor qualified life will be performed prior to the end of the current license period. Although the existing analysis indicates a qualified life of significantly greater than 60 years, the formal reanalysis will verify that the conditions and assumptions used in the original analysis remain valid for the period of extended operation. If the qualification cannot be extended by reanalysis, the motor will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.2.11 Reliance AC Random Wound Motor Model 145TCV

Introduction

The Reliance Model 145TCV motor is used to dewater and prevent flooding of the Auxiliary Building Sub-Basement sump in the event of an RHR pump seal failure during the recirculation phase of a LOCA. The motor was purchased from Reliance in 1995 and qualified to IEEE 323-1974.

Conclusion

The previous thermal and radiation analyses support a qualified life of 40 years. Thermal aging calculations consider the effects of elevated operating temperature during motor operations. Due to the date of installation and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.3 Electrical Penetration Assemblies

4.4.3.1 Crouse-Hinds Electrical Penetration Assemblies

Introduction

Crouse-Hinds Electrical Penetration Assemblies are used for power, control, and instrumentation applications at Ginna station. These penetrations are original plant equipment and qualification is based on DOR Guidelines. The qualification is applicable to the penetration seals, canister, and internal connections. Critical penetration sealing functions are performed by non-organic materials such as ceramic and glass. Electrical conductors internal to Crouse-Hinds penetrations are soldered into ceramic header insulators with splicing to stranded conductor leads accessible at inboard and outboard headers. Additional internal and external electrical insulation is provided by fiberglass and silicone rubber (Varglas), and GE RTV 615.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Self-heating temperature rise was included in the calculation of thermal life. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.3.2 Westinghouse Electrical Penetration Assembly, Model WX32714

Introduction

A single Westinghouse WX32714 electrical penetration assembly was installed at Ginna station in 1975. This penetration is designated for use in instrumentation applications with low voltage and low power requirements. Applications include instrument transmitters, RTDs, and LVDTs. Critical penetration sealing functions are performed by O-rings and Westinghouse Q epoxy. A secondary Scotch epoxy sealant protects the lead connections to the wire electrode feedthroughs. This epoxy surrounds the spliced connections and may be considered the only sealant that is exposed to any effects of beta radiation. This sealant may be considered a sacrificial layer and does not have an adverse impact on the intended function. The primary inner seal is enclosed within a metal tube which protects against beta radiation effects. Qualification is based on DOR guidelines.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Due to the low power circuits, self-heating is not a concern for this application. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.3.3 Westinghouse Modular Electrical Penetration Assemblies

Introduction

Four Westinghouse modular electrical penetration assemblies were installed at Ginna station in 1985. The circuit applications include core exit thermocouples, transmitter circuits, high range radiation monitor, resistance temperature detectors, solenoid operated valves and non-Class 1E welding circuits. The solenoid valves and welding circuits are normally de-energized. All four penetrations are different in internal electrical design, but all use similar materials of construction. These penetrations have a primary inner seal which is enclosed within a metal tube which protects against beta radiation (during accident conditions). A secondary epoxy surrounds the spliced connections, however it may be considered a sacrificial layer and does not have an adverse impact on the intended function. Qualification for these penetrations is performed in accordance with IEEE 323-1976.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Due to the low power circuits, self-heating is not a concern for this application. A review of plant temperatures demonstrated that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.4 Heat Shrink Tubing

4.4.4.1 Raychem WCSF-N Splice Sleeves for Containment Penetration Fan Cooler Motors

Introduction

Raychem WCSF-N Power Cable Sleeves are used in a single EQ penetration application in the Containment Building at Ginna Station. Two sizes of these sleeves are installed as qualified by special test for this unique Ginna Station configuration. These sleeve sizes are Raychem WCSF 600-1250, 300-500 nominal and WCSF1000/3000, 650-1250 MCM nominal. This unique configuration is used to insulate bolted connections between Crouse-Hinds electrical penetration ceramic bushing terminals and power cable circuits supplying Recirculation Fan Cooler motors. The Recirculation Fan Cooler motor circuits are made up of 500 MCM Kerite 600 Volt HTK Insulated, FR Jacketed power cable.

These Raychem WCSF-N Power Cable Sleeves were a one time purchase at Ginna Station made in 1978. Qualification is based on DOR Guidelines.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Self-heating temperature rise was included in the calculation of thermal life. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine the qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.4.2 Raychem Nuclear Splice Kits - NMCK, NPKC, NPKP, and NPKS

Introduction

Raychem Nuclear Splice Kits are used in applications both inside and outside of containment. Splice kits provide an insulation function for electrical conductors. The kits are used for specific applications and are composed of several insulation sleeves. Variations in models and part numbers indicate different sizes and shapes, however all components are constructed of the same Raychem extrusion and molding materials. These Raychem Nuclear Splice Kits were first purchased in 1985. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 44.8 years at full rated temperature. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.4.3 Raychem WCSF-050-N Shim Stock Cable Sleeves

Introduction

Raychem WCSF-050-N is used in applications both inside and outside of containment. Cable sleeves provide an insulation function for electrical conductors. This specific type of insulation sleeve is thin walled and is normally used to build up small diameter cables for use with larger Raychem cable sleeves. These Raychem cable sleeves were first purchased in 1985. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 42.8 years at full rated temperature. Reanalysis of the cable sleeve qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the cable sleeves will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.4.4 Raychem WCSF-N Cable Sleeves

Introduction

Raychem WCSF-N cable sleeves are used in applications both inside and Outside of containment. Cable sleeves provide an insulation function for electrical conductors. The type of cable sleeves covered by this evaluation are general purpose use to environmentally seal cable splices in electrical safety related circuits. These Raychem cable sleeves were first purchased in 1980. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 40 years at full rated temperature. Reanalysis of the cable sleeve qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the cable sleeves will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.4.5 Raychem Nuclear Splice Kits, Model NESK

Introduction

Raychem Nuclear Splice Kits, model NESK are used in applications both inside and Outside of containment. Splice kits provide an insulation function for electrical conductors. These kits are considered cable breakout and end sealing kits used for specific applications. These Raychem Nuclear Splice Kits were first purchased in 1985. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 40 years at full rated temperature. Reanalysis of the splice kit qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the splice kit will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.4.6 Raychem Nuclear Splice Kits, Model NPKV

Introduction

Raychem Nuclear Splice Kits, model NPKV are used in applications both inside and Outside of containment. Splice kits provide an insulation function for electrical conductors. These kits are considered stub connection kits (with end sealing) used for specific applications. These Raychem Nuclear Splice Kits were first purchased in 1982. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 42 years at full rated temperature. Reanalysis of the splice kit qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the splice kit will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.5 Wire and Cable

4.4.5.1 Kerite 600Volt HTK Insulated, FR Jacketed Power Cable

Introduction

Kerite 600Volt HTK Insulated, FR Jacketed Power Cable is used throughout Ginna station, both inside and outside of the Containment Building. EQ applications include the Recirculation Fan Cooler motors, RHR pump motors, Hydrogen Recombiner Fan motors, and high voltage portions of the Hydrogen Recombiner igniter circuits. The electrical loads supplied by these cables do not run continuously. The CRFC and RHR systems use 500MCM and 350MCM cable respectively and are the only EQ applications with a normal run-time of greater than or equal to 50% of service life. Qualification is based on DOR guidelines.

Conclusions

The thermal and radiation reanalyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of cable self-heating. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.2 Kerite 600 Volt FR Insulated, FR Jacketed Control Cable

Introduction

Kerite 600 Volt FR Insulated, FR Jacketed control cable is used throughout Ginna station both inside and outside of the containment building. EQ applications include both inboard and outboard low voltage portions of the Hydrogen Recombiner igniter circuits. Also, inboard control circuits for the residual heat removal discharge motor operated valve (MOV) use this type of cable. Qualification is based on DOR guidelines.

Qualification life for this cable is based on maintaining the electrical function of the insulation. Therefore the insulation is the limiting component for evaluation of cable life.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Normal load current for these control circuits is much less than the ampacity and therefore self-heating is not considered significant. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.3 Conax Electric Conductor Seal Assembly

Introduction

Conax Electrical Conductor Seal Assemblies (ECSAs) are used in the Auxiliary, Containment, and Intermediate Buildings at Ginna Station to seal electrical conductor entrance into electrical terminal housings. An ECSA is a passive electrical device consisting of Kapton polymide film insulated solid copper conductors sealed into a stainless steel tube using polysulfone plastic. The stainless steel tube is swaged into threaded metal fittings. External leads are covered with polyolefin heat shrinkable tubing to provide mechanical protection for the conductor insulation and are routed to terminal boxes via flexible metal conduit.The ECSAs are used to seal solenoid operated valves (SOVs) and resistance temperature detectors (RTDs). Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of elevated operating temperature during SOV operation. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.4 Anaconda FR-EP and FR-EP/CPE Cable

Introduction

Anaconda FR-EP and FR-EP/CPE (CPE Jacket) instrument and control cable is used throughout Ginna station in motor operated vale (MOV), solenoid operated valve (SOV), position switch/indicator, thermocouple, pressure transmitter, and resistance temperature detector applications. The cable is installed both inside and outside the Containment Building. The MOV power applications use #10 AWG field wiring extending from penetration splice boxes to the pressurizer cubicle. The stroke time for these valves is less than two minutes. Therefore operation is a negligible portion of service life and self-heating due to power loads is not a significant contributor to thermal aging. The MOV and position switch applications use #14 AWG cable for internal DC control wiring and #12 AWG for position switch wiring, respectively. Qualification is based on IEEE 323-1974. A unique (and limiting) application of this cable is in the pressurizer cubicle. The temperature inside of the cubicle is significantly higher than ambient.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Normal load current for these circuits is much less than the ampacity and therefore self-heating is not considered significant. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.5 General PVC Insulated and Jacketed Control Cable

Introduction

General PVC insulated and jacketed control cable is used outside of the containment building for two EQ solenoid operated valves that are associated with the Post Accident Sampling System. The SOV applications use #12 multi-conductor cables installed between a control panel and a penetration outboard splice box. Qualification is consistent with DOR guidelines.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Normal loading on this type of cable consists of position indication lamps which present a load current that is much less than the cable ampacity. Therefore, self-heating is not considered significant. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation, and no reductions in excess conservatisms were necessary. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.6 PVC Instrument Cable Outside Containment (Rome Cable Corp.)

Introduction

Rome Cable Corporation PVC Insulated and Jacketed Instrument cable is used outside the Containment Building in Hydrogen Recombiner circuits running between the control panels and Intermediate Building outboard penetration splice boxes. These circuits carry flow transmitter and thermocouple signals. This cable is also used in pressure transmitter circuits located in both the Auxiliary and Intermediate Buildings. These Auxiliary and Intermediate Building circuits are subject to HELB environments. The conductors are PVC insulated, twisted, shielded and jacketed with glass braid. Qualification is based on a combination of tests which are acceptable for DOR Guideline equipment.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. All of the EQ cables carry low voltage level analog instrument signals and operate with milliampere circuit loadings and therefore have negligible cable self-heating. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models, temperature, and activation energies used in the reanalysis are the same as those used in the prior evaluation, and no reductions in excess conservatisms were necessary. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.7 BIW Tefzel ETFE Coaxial and Triaxial Cable

Introduction

BIW Tefzel ETFE Coaxial and Triaxial Cable are used in two High Range Radiation Monitor installations at Ginna Station. The High Range Radiation Monitor detectors are located in Containment. There are two circuits for each detector; these are separate high voltage supply and signal cables. For each circuit BIW Tefzel ETFE coaxial cable is run in Containment and the Intermediate Building. BIW Tefzel ETFE Coaxial and Triaxial Cable installed at Ginna Station is qualified to IEEE 323-1974. The BIW coaxial cable was purchased from Victoreen with the High Range Radiation Monitors in 1979. The triaxial cable was purchased from BIW in 1986. Both the cable insulation and jacket material is Dupont Tefzel fluoropolymer (ETFE). These cables operate with milliampere circuit loadings and therefore have negligible cable self-heating.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 50 years. These cables operate with milliampere circuit loadings and therefore have negligible cable self-heating. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Since the purchase order date for the HRRMs including coaxial cable is 1979 and the triaxial cable was purchased at a later date, this qualified life provides adequate margin for extended operation to 2029. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.8 BIW EPR Insulated and CSPE Jacketed Cable

Introduction

BIW EPR Insulated and CSPE Jacketed cable is used inside the Containment Building and throughout the Ginna StationGinna Station. EQ applications in the Containment Building include motor operated valve (MOV) control, solenoid operated valve (SOV), position indication, pressure transmitter, and resistance temperature detector circuits. MOV control circuits extend from containment electrical penetrations to the pressurizer cubicle.. A bounding analysis was performed by applying the pressurizer cubicle temperatures to the SOV circuits.

Cable loading for the SOVs is based on twice the normal operating requirements of once per week or 52 cycles per year at one hour per cycle. This results in an average of 104 operating hours per year. This cable was first purchased for installation in EQ applications at Ginna Station in 1985. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Thermal aging calculations consider the effects of cable self-heating. A review of plant temperatures in the reanalysis provide reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.9 Raychem Flamtrol Shielded Cable

Introduction

Raychem Flamtrol Shielded cable is used in the Containment Building at Ginna Station in EQ position indication, and pressure transmitter applications. Power Operated Relief Valve position indicator circuits extend in conduit from the Containment Building basement up the pressurizer cubicle inside walls to the top level. This cable was a one time purchase at Ginna Station made in 1975. The elevated ambient temperature in the pressurizer cubicle is considered the worst case for this installation. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. All EQ Raychem Flamtrol Shielded cable circuits are instrument applications, operate with milliampere circuit loadings, and therefore have negligible cable self-heating. A review of plant temperatures demonstrates that the temperature used to determine the qualified life is conservative. The analytical models, temperatures, and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.10 General PVC Insulated and Jacketed Control Cable

Introduction

General PVC Insulated and Jacketed Control cable is used inside the Containment Building at Ginna Station in a single EQ solenoid operated valve (SOV) application . This SOV controls a Post Accident Sample System sample valve located in the Containment Building basement, Loop B area. This SOV application uses both single conductor #14 AWG and ten conductor #12 AWG cable. The operating function requires energization 1.5 times per day for a total energized time of 45 minutes per day. This cable is original plant equipment. The elevated ambient temperature in the pressurizer cubicle is considered the worst case for this installation. Qualification is based DOR Guidelines.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Normal load current for this control cable is much less than the ampacity and therefore self-heating is not considered significant. A review of plant temperatures in the reanalysis provide reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.11 Containment Electrical Penetration Pigtail Extension Cables

Introduction

The penetration extension cable is single conductor silicone rubber insulated with a braided jacket. This cable is noted as installed in containment. Qualification is based on DOR guidelines.

Conclusion

The existing thermal and radiation analyses support a qualified life of 52 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life remains conservative. At this time, there are no plans to extend the qualified life of the existing cable, and therefore the cable will remain scheduled for replacement prior to the year 2021. In the event that reanalysis is performed, it will be completed consistent with the guidance provided in NUREG-1800 table 4-4. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.5.12 Coleman Silicone Rubber Instrument Cable Inside Containment

Introduction

The Coleman cable is twisted shielded instrumentation cable with silicone rubber insulation and a glass braid jacket. This cable is noted as installed in containment. Qualification is based on DOR guidelines.

Conclusion

The previous therman and radiation analyses support a qualified life of 52 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. At this time, there are no plans to extend the qualified life of the existing cable, and therefore the cable will remain scheduled for replacement prior to the year 2021. In the event that reanalysis is performed, it will be completed consistent with the guidance provided in NUREG-1800 table 4-4. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.5.13 Okonite Control Cable

Introduction

The Okonite cable which is part of the Reactor Head Vent cable-connector assemblies is 5/C #12 with 0.30 Okonite-FMR (EP) insulation and 0.045 Okolon (SCPE) jacket. This cable is an integral part of the Reactor Head Vent cable connector assembly and therefore the qualified life is considered to be limited by the connector qualified life (30 years). Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 30 years. A review of plant temperatures for the existing analyses provide reasonable assurance that the temperature used to determine qualified life remains conservative. At this time, there are no plans to extend the qualified life of the existing cable-connector assemblies, and therefore these assemblies will remain scheduled for replacement prior to the year 2017. In the event that reanalysis is performed, it will be completed consistent with the guidance provided in NUREG-1800 table 4-4. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.5.14 Conax Core Exit Thermocouple Connector/Cable Assemblies

Introduction

Conax Core Exit Thermocouple (CET) Connector/Cable assemblies are installed in the Containment Building, in open trays above and to the side of the reactor head. These Connector/Cable assemblies provide signals for display of Reactor Coolant System temperature in the reactor vessel. The Conax CET Connector/Cable assemblies are circular two pin connectors consisting of a mechanically swaged stainless steel, polysulfone and Kapton wire insulation interfaces. The oldest of these assemblies was purchased in 1984 with installation that year.

For the evaluation of qualified life, the polysulfone insulating material used in the connector is the limiting component for the thermal analysis, and the Kapton wire insulation in the connector is the limiting component for the radiation analysis.

Conclusion

The thermal, radiation and cyclical wear analyses support a qualified life in excess of 60 years. Normal load current for these Connector/Cable assemblies is much less than the ampacity and therefore self-heating is not considered significant. A review of plant temperatures in the reanalysis provides reasonable assurance that the temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.5.15 Namco Limit Switch Connector/Cable Assemblies

Introduction

The Namco limit switch connector/cable assemblies are used for pressurizer PORV position indication. The connector/cable assemblies were purchased in 1994 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. The normal load current through the pressurizer PORV limit switches is much less than the ampacity and therefore self-heating of the connector/cable assemblies is not considered significant. A review of plant temperatures in the existing analysis provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.5.16 Brand Rex Electrical Cable

Introduction

The Brand Rex cables are used to provide instrumentation and control power for class 1E circuits. These cables were purchased from 1985 to 1997 and qualified to IEEE 323-1974.

Conclusion

The existing thermal and radiation analyses support a qualified life of 40 years. Reanalysis of the cable qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the cable will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.6 Electrical Connectors

4.4.6.1 Amphenol Triaxial Cable Connector

Introduction

Amphenol Triaxial Cable Connectors (Plug#53175-1004 and Jack#52957-1001) are used in the high range radiation monitor circuits. The connectors were purchased in 1989 and qualified to IEEE 323-1974. The insulator used in the connector is made of cross linked polyolefin.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. The installed application is for instrumentation circuits and therefore self-heating temperature rise is not considered to be significant. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.6.2 EGS Quick Disconnect Electrical Connectors, ITT Cannon GB-1

Introduction

ITT Cannon GB-1 connectors supplied by EGS are used to provide environmentally sealed connections for Class 1E circuits. The connector is a single conductor, separable connector with an elastomer body. The connectors were purchased in 1992 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.6.3 Quick Disconnect Electrical Connectors, EGS

Introduction

EGS quick disconnect connectors are used to provide environmentally sealed connections for Class 1E power and control circuits. These are multiple conductor, bayonet type, quick-disconnect connectors. The connectors were purchased in 1996 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.6.4 States Terminal Blocks Model M-25012

Introduction

States terminal blocks are used for an EQ application in the Intermediate Building for main steam line isolation valve instrument air solenoid valves. The terminal block is a passive device consisting of phenolic as the most age sensitive material. Plant documents identify the States terminal blocks were installed in 1983. The installed location is not normally considered a harsh environment, however the equipment supported must maintain the intended function for a main steam line break scenario. Therefore the components are considered to be exposed to elevated temperatures for a portion of the service life which will encompass the design basis event. Qualification is based on DOR guidelines.

Conclusion

The thermal and radiation analysis support a qualifies life of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Reanalysis of the terminal block qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the terminal blocks will be refurbished, replaced, or requalified prior to exceeding the current qualification.

Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.6.5 AMP Butt Splices, Models 53549-1 and 53550-1

Introduction

AMP Nuclear PIDG Window Indent Butt Splices Models 53549-1 and 53550-1, are used in containment and throughout the plant for electrical connections. The butt splices are a passive component, consisting of KYNAR Polyvinylidene Fluoride (PVDF) as the most age sensitive material. Plant documents identify that the AMP butt splices were first purchased for EQ applications in 1985 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Reanalysis of the splice qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the splice will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.6.6 AMP Butt Splices, Models 52979 and 52980

Introduction

AMP Nuclear pre-insulated environmentally sealed butt splices Models 52979 and 52980, are used in containment and throughout the plant for electrical connections. The butt splices are a passive component, consisting of KYNAR Polyvinylidene Fluoride (PVDF) as the most age sensitive material. Plant documents identify that the AMP butt splices were first purchased for EQ applications in 1985 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. The qualified life of these components considers the self-heating temperature rise consistent with rated load current. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Reanalysis of the splice qualified life will be performed prior to the end of the current license period. If the qualification cannot be extended by reanalysis, the splice will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.4.7 Resistance Temperature Detectors

4.4.7.1 Conax Resistance Temperature Detector Models 7N92-10000 and 7A22-10000

Introduction

Conax RTD Models 7N92-10000 and 7A22-10000 are used in the reactor coolant system to measure process fluid temperatures.These units are direct immersion RTDs located in the A and B hot and cold leg piping, inside the shield wall with cold end lead configurations external to process piping insulation. The two models account for two different cold end lead configurations.

These RTDs consist of dual three wire platinum coils connecting to Kapton polyimide film insulated solid copper leads all in a mineral insulated stainless steel sheath with polysulfone plastic sealant at the cold end lead wire exit. Mineral electrical insulation is used where hot end components are exposed to process heating.

Plant documents identify the oldest of the installed RTDs to have been purchased in 1982. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Self heating for instrument cable is considered to be insignificant. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.7.2 Conax Resistance Temperature Detector Model 7DB9-10000

Introduction

Conax RTD Model 7DB9-10000 is used in the reactor vessel level monitoring system to measure process fluid temperatures. These submersible RTDs are installed in three locations outside the shield wall. These RTDs consist of dual three wire platinum coils connecting to Kapton polyimide film insulated solid copper leads all in a mineral insulated stainless steel sheath with polysulfone plastic sealant at the cold end lead wire exit. Self heating for instrument cable is considered to be insignificant. Mineral electrical insulation is used where hot end components are exposed to process heating. The effects of elevated ambient temperatures have been considered for the calculation of thermal life.

Plant documents identify the installed RTDs were purchased and installed in 1989. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. Self heating for instrument cable is considered to be insignificant. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.7.3 Pyromation Resistance Temperature Detector

Introduction

Pyromation model RT186S28123-00-36Z RTDs are used to provide Containment Building atmosphere temperature monitoring. This is an indicating function based on Regulatory Guide 1.97 Category II. These RTDs consist of a single three wire platinum coil connecting to Kapton polyimide film insulated solid copper leads. The limiting age degradable material is the United Resins potting compound used for lead wire terminations. The RTDs were first purchased in 1989 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40.5 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.8 Victoreen High Range Radiation Monitor

Introduction

Two Victoreen supplied Detector and Cable/Connector assemblies are installed in the Containment Building near the outside wall. . These components are used with Victoreen High Range Radiation Monitors which are located in the Control Room. The assemblies were installed with their current connector configuration in 1982 and were originally qualified to the DOR Guidelines.

Detector materials are metals and ceramics, with no applicable aging mechanisms. Qualification life is therefore based on the cable/connector assembly. For the evaluation of qualified life, the cable insulation (Tefzel) is the limiting component for the thermal analysis, and the Raychem connector encapsulation seal is the limiting component for the radiation analysis.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. There is no significant self-heating associated with instrument and control wiring. A review of plant temperatures provides reasonable assurance that the reduced temperature used to determine qualified life is conservative. The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.9 Rosemount Conduit Seal

Introduction

Rosemount conduit seals (model 353C) are used to seal transmitter housings and transmit process signals for steam generator wide range level indication. The seals were purchased in 1991 and qualified to IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life of 40 years. A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. Due to the date of purchase and the existing qualified life, reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.10 Transamerica Delaval Level Switch

Introduction

The Transamerica Delaval, Gem Level Sensor , Model LS57761 is a float switch used in the Containment Building basement to provide Sump B level indication following a design basis event. The level sensor is a magnetic reed type liquid switch. A spherical float surrounds a stationary stem and moved up and down with the liquid level. All metal parts are stainless steel and the stem and junction box are completely filled with Dow-Corning 710 silicone fluid, with splices submerged. The junction box is connected to conduit in a configuration arranged to avoid field wiring submergence prior to switch activation. Qualification is performed in accordance with IEEE 323-1974.

Conclusion

The thermal and radiation analyses support a qualified life in excess of 60 years. A review of plant temperatures in the reanalysis provides reasonable assurance that the reduced temperature used to determine qualified life is conservative.The analytical models and activation energies used in the reanalysis are the same as those used in the prior evaluation, and no reductions in excess conservatisims were necessary. Based on this review, the analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.4.11 Schaevitz Linear Variable Differential Transformer

Introduction

Schaevitz Linear Variable Differential Transformers (LVDTs) Model 500 XS-ZTR are used for indication of pressurizer safety valve position. The LVDTs were purchased in 1986 and qualified based on DOR guidelines. They are designed to operate continuously under 2500 psi, 650EF conditions, with a total integrated dose of 2.5E11 RADs. These design conditions are much more severe than the postulated accident conditions at Ginna.

Conclusion

A review of plant temperatures provides reasonable assurance that the temperature used to determine qualified life is conservative. The LVDTs are constructed of completely inorganic insulating materials (i.e. ceramics and metals). Therefore, there are no aging effects in the period of extended operation, and reanalysis is not required for license renewal purposes. The existing analyses remain valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.4.12 Hydrogen Recombiner Exciter, Ignitor, and Thermocouples

Introduction

The Hydrogen Recombiner system consists of two full-rated subsystems, each capable of maintaining the post-LOCA containment hydrogen concentration below 4 volume percent. The specific subsystem components covered by this evaluation include the exciter, ignitor, and thermcouples. The exciter consists of a transformer, capacitors, inductor, rectifiers, electron discharge tube and resistors enclosed in a seal welded metal case. The input and output connections are made with a gasket sealed junction box welded to the side of the metal case. The junction box is filled with GE RTV-7403 to preclude the entry of moisture. The ignitor is a component that produces a spark from the 2300V DC pulse generated by the exciter. This component does not contain any sub-components which are susceptible to age, radiation, temperature or chemical spray degradation. To ensure proper lite-off and temperature control of the hydrogen recombiners, they incorporate two thermocouples that sense the combustor outlet temperature. Qualification for this equipment is based on DOR guidelines.

Conclusion

Reanalysis of the hydrogen recombiner equipment qualified life will be performed prior to the end of the current license period. If the qualification is still required by 10 CFR 50.44, and cannot be extended by reanalysis, the components will be refurbished, replaced, or requalified prior to exceeding the current qualification. Therefore the existing environmental qualification process, in accordance with 10 CFR 50.49, will adequately manage aging of EQ equipment for the period of extended operation. This is consistent with 10 CFR 54.21(c)(1)(iii).

4.5 Concrete Containment Tendon Prestress

Introduction

The Ginna Station containment structure is a reinforced concrete, vertical right cylinder with a flat base and hemispherical dome. The cylinder wall is prestressed vertically and reinforced circumferentially with mild steel deformed bars. Vertical prestress is provided by 160 unbonded post-tensioned tendons. The base slab and dome are constructed of reinforced concrete.

The design of the containment provides for prestressing the cylinder walls in the vertical direction with sufficient compressive force to ensure that for all design load combinations there are no membrane tensile forces in the concrete. The design requires that all bending and shear forces be resisted by mild steel reinforcement, which controls potential crack width, spacing, and depth.

The prestressing force of containment tendons may decrease over time due to creep, shrinkage and elastic shortening of the concrete, and stress-relaxation of the prestressing tendon wires. Prestressing tendon integrity is monitored and confirmed by the ASME Section XI, Subsection IWE/IWL Inservice Inspection Program.

An analysis was performed to evaluate the trend in the loss of prestress for each of the 160 tendons at Ginna Station. A review of the historical lift-off force measurements for the tendons was conducted. It was appropriate to review the results as two separate groups, i.e., the 23 tendons which were retensioned in 1969, and the 137 tendons which were retensioned in 1980. Of the 23 tendons that were retensioned in 1969, eleven have been tested during the surveillances since 1980. Of the 137 tendons that were retensioned in 1980, forty-seven have been tested during the subsequent surveillances. The number of tendons sampled during the surveillance tests exceeds the requirements of Regulatory Guide 1.35.

Using the guidance in RG 1.35.1, tolerance bands were calculated and the lift-off forces measured during surveillance tests were expressed in terms of margins. It was concluded that the group of 23 tendons originally retensioned in 1969 should be retensioned as documented in the Evaluation of Loss of Prestress in Containment Tendons TLAA . These tendons have exhibited loss of prestress as determined during previous surveillance tests. Retensioning should preclude further loss of prestress.

Conclusion

It is concluded that retensioning the group of 23 tendons in 2005 will provide additional assurance that the minimum design tendon prestress force will be maintained through the period of extended operation.

Based on this review, the ASME Section XI, Subsections IWE & IWL Inservice Inspection Program will manage the effects of aging for the Containment post-tensioning system during the extended period of operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.6 Containment Liner Plate and Penetration Fatigue

Introduction

The interior surface of the Containment Structure is lined with welded steel plate to provide an essentially leak-tight barrier. At all penetrations, the liner plate is thickened to reduce stress concentrations.

The containment liner, liner penetrations and liner steel components of the Ginna Station Containment Structure comply with the ASME Code Section III-1965 for pressure boundary and the AISC Code for structural steel. The containment liner and penetrations, including the equipment and personnel hatch penetrations, were designed as Class B Vessels. The Winter 1965 Addenda of ASME Section III, Subsection B, N-1314(a) requires that the containment vessel satisfy the provisions of Subsection A, N-415.1, "Vessels Not Requiring Analysis for Cyclic Operation," in order that Subsection B rules be applicable.

ASME 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 (6) conditions. These conditions address the following types of loads:

- Atmospheric to Service Pressure Cycles
- Normal Service Pressure Fluctuation
- Temperature Difference - Startup and Shutdown
- Temperature Difference - Normal Service
- Temperature Difference - Dissimilar Materials
- Mechanical Loads

The pressure boundary components analyzed include the liner adjacent to the penetration, the penetration sleeve, and the annular plate connecting the pressure piping to the sleeve.

An analysis was performed which verifies that each of the six conditions described above are satisfied for the period of extended operation. This analysis demonstrates that the liner and penetrations comply with the ASME Section III - 1965 Code Rules for fatigue through the period of extended operation.

Conclusion

The six criteria of ASME Code Section III, N-415.1 (Reference 17) have been reevaluated and shown to be satisfied for the Containment liner plate, penetrations and penetration sleeves for 60 years of plant operation. Based on this evaluation, the fatigue analyses have been projected to the end of the extended period of operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.7 Other Plant Specific TLAAs

4.7.1 Containment Liner Stress

Introduction

The containment liner is carbon steel plate conforming to ASTM A442-60T Grade 60 with a minimum yield of 32,000 psi, and a buckling stress of 16,600 psi at operating conditions. The liner plate thickness is 1/4 in. for the base and 3/8 in. for the cylinder and dome. The liner stresses (meridional directions) were calculated to be 4500 psi compression based upon a prestress force of 0.70 fs. The concrete strain due to creep and shrinkage was established as being 320 x 10^{-6} in/in. This increases the liner stress to 14,100 psi at the end of 40 years.

Conclusion

The creep and shrinkage strain occurring over a 60-year plant life was calculated, and the resulting compressive liner stress due to both time-dependent and non-time dependent loads is determined to be 14,870 psi. This liner stress is less than the liner buckling stress of 16,600 psi and therefore the analysis has been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

4.7.2 Containment Tendon Fatigue

Introduction

A discussion of seismic considerations for tendons is provided in the Ginna UFSAR. Fatigue tests were conducted on tendon wire materials in 1960 by an independent testing lab. The tests indicated that the tendons were capable of withstanding over 2 million cycles at stress levels between 135 and 158 ksi. The test results were used to conclude that dynamic loads, considering especially pulsating loads resulting from an earthquake, do not jeopardize buttonhead anchorage.

This discussion may not meet the definition of a TLAA as described in 10 CFR 54.3, however it has been included for conservatism.

Conclusion

The tendon fatigue test results described in the UFSAR were cited to address tendon integrity during cyclic loading of the containment caused by a design basis seismic event. The tendons were tested to 2 million cycles, which exceeds, by many orders of magnitude, the total cycles that could accumulate through multiple seismic events over 60 years. The seismic fatigue evaluation remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.7.3 Containment Liner Anchorage Fatigue

Introduction

A fatigue analysis of the fillet weld attaching the channel anchors to the liner was performed as part of the original 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 corresponds to 100,000 stress cycles.

This discussion may not meet the definition of a TLAA as described in 10 CFR 54.3, however it has been included for conservatism.

Conclusion

A total of 100,000 stress cycles corresponds to more than 4 full stress cycles per day for 60 years. Fluctuations of temperature and pressure in containment on a daily basis are not significant enough in magnitude to cause four cycles of design basis stress at the liner anchorage weld each day. Therefore, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.7.4 Containment Tendon Bellows Fatigue

Introduction

The allowable radial and vertical displacements of the containment stainless steel tendon bellows are given in the UFSAR 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. Since the completion of construction, displacements at the tendon bellows have occurred due to pressure testing and temperature changes in the cylindrical shell wall due to summer/winter conditions and reactor shutdown during refueling outages.

This discussion may not meet the definition of a TLAA as described in 10 CFR 54.3, however it has been included for conservatism.

Conclusion

Assuming that 80 full cycles of allowable displacement results in a fatigue usage factor of 1.0, the actual fatigue usage factor over a 60-year period has been calculated to be much less than 0.01. Therefore, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

4.7.5 Crane Load Cycle Limit

Introduction

Each of the crane estimated cycle numbers were compared to the Design Load Cycles. They are all well below the upper Design Loading Cycle limit. In addition, the average percent of the rated load lifted was well below the 50% level, relative to the design load cycles, as set forth in the design criteria.

Conclusion

Since the number of operating load cycles for the cranes will be less than the design cycles and the average percent of rated load lifted is less than 50% for the design load cycles, the crane designs will remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

4.7.6 RCP Flywheel

Introduction

During normal operation, the reactor coolant pump (RCP) flywheel possesses sufficient kinetic energy to produce high-energy missiles in the event of failure. Conditions which may result in over-speed of the RCP increase both the potential for failure and the kinetic energy. The aging effect of concern is fatigue crack growth in the flywheel bore keyway.

Westinghouse Topical Report WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination" (Reference 18), presents an evaluation of the probability of failure over an extended operating period of 60 years. This report demonstrates that the flywheel design has a high structural reliability with very high flaw tolerance and negligible flaw crack extension over a 60-year service life. The Westinghouse Topical Report provides technical justification for elimination of the RCP flywheel inspection. This Topical report was reviewed and approved by the NRC (Reference 19), for referencing in licensing, with certain restrictions and limitations specified. This was re-published as WCAP-14535A in November 1996.

Regulatory Position C.4.b(1) of NRC Regulatory Guide 1.14, Revision 1, August 1975 (Reference 20), recommended performance of an in-place ultrasonic volumetric examination of the areas of higher stress concentration at the Reactor Coolant Pump flywheel bore and keyway at approximately 3-year intervals. Regulatory Position C.4.b(2) of this regulatory guide (Reference 20), recommended a surface examination of all exposed surfaces and complete ultrasonic examination at approximately 10-year intervals, coinciding with the ASME Section XI Inservice Inspection program schedule. This recommendation was incorporated in the Ginna Station Inservice Inspection (ISI) program. Based on

WCAP14535A, and in accordance with NRC recommendations, RG&E requested and received a releif request from the NRC revising the ISI frequency of examination to perform the flywheel inspection once every 10 years. The method of examination includes either an ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or an ultrasonic and a surface examination of exposed surfaces defined by the volume of the disassembled flywheel.

Conclusion

In accordance with the requirements of the ISI Program, the RCP flywheel inspection program will continue to ensure that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii).

4.7.7 Thermal Aging of Cast Austenitic Stainless Steel (CASS)

Introduction

Thermal aging refers to changes in the microstructure and properties of a susceptible material due to prolonged exposure to elevated temperatures above 482°F. Typical RCS temperatures exceed this threshold. The effect of thermal aging on Class 1 components is loss of fracture toughness (embrittlement) of the duplex ferritic-austenitic stainless steel elbow castings in the reactor coolant piping and the cast reactor coolant pump casings. Since the embrittlement reaction is time-dependent, the associated aging effect is therefore treated as a TLAA. A Leak-Before-Break (LBB) (flaw tolerance) analysis is typically performed to demonstrate that any leaks from through-wall cracks that develop in RCS piping would be detected by plant monitoring systems before the cracks could grow to unstable proportions. For RCS piping, this analysis must consider the reduction in fracture toughness of CASS as a result of thermal aging.

A fracture mechanics analysis (Reference 27) has been performed which considers loading, pipe geometry and fracture toughness to assess crack stability in the reactor coolant piping for the period of extended operation. This analysis, which considered the reduction of fracture toughness in CASS elbows in the RCS piping for the period of extended operation, again demonstrated that significant margin exists between detectable flaw sizes and unstable flaws. Additionally, fatigue crack growth rates including environmental effects were evaluated for primary loop piping and shown to be insignificant.

Similarly, in lieu of performing volumetric inspections of the cast austenitic stainless steel (CF8M) RCP casings, a fracture mechanics analysis, according to the requirements of ASME Code Case N-481 has been performed for the period of extended operation (Reference 28). The results of this analysis demonstrated that the fracture toughness of the pump casing materials in the fully-aged condition is sufficient to meet the stability criteria for a postulated flaw.

Conclusions

Flaw-tolerance analyses have been performed to evaluate the reduction in fracture toughness due to thermal aging of CASS reactor coolant system elbows and the RCP casings through the extended period of operation. The results demonstrate that large margins exist for postulated flaw sizes against flaw instability. Therefore, the analyses have been projected to the end of the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

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APPENDIX A

UFSAR SUPPLEMENT

APPENDIX A
Contents

Contents

A1.0 APPENDIX A - INTRODUCTION

The application for a renewed operating license is required by 10 CFR 54.21(d) to include a FSAR Supplement. This appendix provides that supplement for the Ginna UFSAR. Section A2.0 of this appendix contains a summarized description of the programs for managing the effects of aging. Section A3.0 of this appendix contains a summary of the evaluation of time-limited aging analyses (TLAAs) for the period of extended operation. TLAA supporting activity summaries are contained in Section A4.0

A2.0 PROGRAMS THAT MANAGE THE EFFECTS OF AGING

This section provides summaries of the programs and activities credited for managing the effects of aging, in alphabetical order. The Ginna Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Section A.2 of NUREG-1800, Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, published July 2001. The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and non- safety-related structures, systems, and components that are within the scope of license renewal.

A2.1 Aging Management Programs

The description of the Ginna Aging Management Programs are consistent with their status as configured to apply to the period of extended operation.

A2.1.1 Aboveground Carbon Steel Tanks

The functional intent of this program is implemented by the Systems Monitoring and One-Time Inspection Programs. The programs provide for periodic system walkdowns and inspections to monitor the condition of selected above ground carbon steel storage tanks, including an assessment of tank surfaces protected by paints or coatings, although the coatings themselves are not credited to perform a preventive function. For inaccessible surfaces such as concrete foundation interfaces, an inspection of the tank bottom wall thickness is performed.

A2.1.2 ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection

The program consists of periodic volumetric, surface, and/or visual examinations and leakage tests of all Class 1, 2, and 3 pressure-retaining components to identify evidence of degradation. Surface and visual examinations of integral attachments are also performed. This program is in accordance with ASME Section XI, 1995 edition through the 1996 addenda. The program also provides for evaluation of inspection results and appropriate corrective actions.

A2.1.3 ASME Section XI, Subsections IWE & IWL Inservice Inspection

The program consists of periodic visual inspection of concrete surfaces for reinforced and prestressed concrete containments, and periodic visual inspection and sample tendon testing of unbonded post-tensioning systems for prestressed concrete containments, for evidence of degradation, assessment of damage and corrective actions. Measured tendon lift-off forces are compared to predicted tendon forces calculated in accordance with Regulatory Guide 1.35. The ASME Section XI, Subsection IWE program consists of periodic visual, surface, and volumetric inspection of pressure retaining components of steel and concrete containments for evidence of degradation. The program also provides for assessment of damage and appropriate corrective actions. This program is in accordance with ASME Section XI, Subsections IWE and IWL, 1992 edition including 1992 addenda.

A2.1.4 ASME Section XI, Subsection IWF Inservice Inspection

This program consists of periodic visual examinations of component supports for evidence of degradation. The program provides for evaluation of inspection results and appropriate corrective actions. This program is in accordance with ASME Section XI, Subsection IWF, 1995 edition, including 1996 addenda.

A2.1.5 Bolting Integrity

The functional intent of this program is implemented by the following programs: 1) ASME Section XI, Subsections IWB, IWC, and IWD ISI Program, 2) ASME Section Subsection IWF ISI Program, 3) Periodic Surveillance and Preventive Maintenance Program, 4) Boric Acid Corrosion Program, 5) Systems Monitoring Program, and 6) Structures Monitoring Program. The program consists of periodic inspections of pressure retaining bolting as delineated in NUREG-1339, and other industry recommendations in EPRI NP-5679 (with exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for pressure retaining and structural bolting. The program provides for periodic inspection of closure bolting for indication of loss of preload, cracking, and loss of material.

A2.1.6 Boric Acid Corrosion

The program consists of: (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of any damage, and (4) follow-up inspections to assure effectiveness of corrective actions. The program scope includes RCS components in accordance with Generic Letter 88-05 as well as non-RCS mechanical, electrical and structural components susceptible to boric acid corrosion which are potentially exposed to borated water leaks.

A2.1.7 Buried Piping and Tanks Inspection

The functional intent of this program is implemented by the One-Time Inspection Program. The Program includes provisions for: (1) preventive measures to mitigate corrosion, and (2) periodic inspections to manage the effects of corrosion on the pressure retaining capacity of buried carbon steel piping and tanks during inspections of opportunity. Preventive measures are in accordance with standard industry practice for maintaining external coatings and wrappings. Buried piping and tanks are inspected visually for any evidence of degradation when they are uncovered for any reason.

A2.1.8 Closed-Cycle (Component) Cooling Water System

The program includes: 1) preventive measures to minimize corrosion by maintaining corrosion inhibitor concentrations within specified limits, 2) surveillance tests and inspections, and 3) nondestructive evaluations of internal surfaces of system components. Evaluations to verify the effectiveness of water chemistry controls are based on the guidelines of EPRI TR-107396 for closed-cycle cooling water systems.

A2.1.9 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The program requires that cables and connections in accessible areas exposed to adverse localized environments caused by heat, radiation, or moisture are inspected on a periodic basis. Visual inspections for cable and connector jacket surface anomalies such as embrittlement, discoloration, cracking, and surface contamination are performed at least once every ten years.

A2.1.10 Fire Protection

The program includes inspection of fire barriers and functional testing of fire pumps. The Fire Protection Program requires periodic visual inspections of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspections and functional tests of fire rated doors and dampers to ensure that their operability is maintained. The Fire Protection Program requires that the diesel driven fire pump be periodically tested to ensure that the fuel supply line can perform the intended function. The program also includes periodic inspection and testing of the halon fire suppression system.

A2.1.11 Fire Water System

The program consists of inspections and functional tests of fire suppression components such as sprinklers, hydrants, valves and piping. Periodic full flow flush tests and system performance tests are conducted to prevent corrosion due to silting and biofouling of components. In addition, the system is normally maintained at required operating pressure and is monitored such that loss of system pressure is immediately detected and corrective actions initiated. Internal portions of the fire water system are visually inspected when disassembled for maintenance. Volumetric NDE inspections using appropriate techniques are performed to detect wall loss and fouling. Replacement or representative sample testing of sprinklers with a service life of 50 years is specified.

A2.1.12 Flow-Accelerated Corrosion

The program consists of: (1) conducting appropriate analyses to determine critical locations susceptible to FAC, (2) conducting baseline inspections to determine the extent of thinning at these locations, and (3) performing follow-up inspections to confirm predicted degradation rates. Corrective actions such as repair or replacement are evaluated based on inspection results and predicted rates of wall loss. The program implements the EPRI guidelines in the Nuclear Safety Analysis Center (NSAC) 202L-R2 and utilizes the CHECWORKS predictive code.

A2.1.13 Fuel Oil Chemistry

The program consists of a combination of surveillance and maintenance activities. Monitoring and control of fuel oil contamination in accordance with the guidelines in ASTM Standards D975, D1796, D2709 and D4057 maintains fuel oil quality. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by periodic cleaning/draining of storage tanks and verifying the quality of new fuel oil before introduction into the tanks.

A2.1.14 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

The program evaluates the effectiveness of testing and monitoring activities as well as the effects of past and future usage on the structural reliability of the cranes, hoists and lifting devices that were evaluated in Ginna Station's response to NUREG-0612. The number and magnitude of lifts made by the hoist or crane are also reviewed. Rails and girders are visually inspected on a periodic basis for evidence of degradation. Functional tests are also performed to assure structural integrity.

A2.1.15 One-Time Inspection

The intent of this program is to verify the effectiveness of existing aging management programs by confirming the absence of an aging effect or verifying that the aging effect is developing so slowly that the intended function is expected to be maintained through the period of extended operation. The program methodology includes selection of appropriate inspection techniques and sample size to ensure that the specified age-related degradation will be discovered in a timely manner. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.16 Open-Cycle Cooling (Service) Water System

The program is based on implementation of the recommendations of Generic Letter 89-13 to ensure that the effects of aging on open cycle cooling water system components will be managed for the extended period of operation. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the open cycle cooling water system or structures and components serviced by the open cycle cooling water system.

A2.1.17 Periodic Surveillance and Preventive Maintenance

The Periodic Surveillance and Preventive Maintenance Program is used to maintain plant equipment and structure condition and ensure reliability at Ginna Station. The program provides for visual inspection and examination of surfaces of selected equipment items and components, including fasteners, for evidence age-related degradation such as corrosion, wear, cracking, fouling, and loss of mechanical closure integrity on a specified frequency based on operating experience and previous inspection results. The program provides for evaluation of inspection results and appropriate corrective actions. The program also provides for replacement or refurbishment of certain components on a specified frequency based on operational experience.

A2.1.18 Reactor Vessel Head Penetration Inspection

The Reactor Vessel Head Penetration Inspection program includes: 1) susceptibility assessment of head components (including alloy 690TT subcomponents) to primary water stress corrosion cracking (PWSCC), 2) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and 3) inservice inspection (ISI) of reactor vessel head penetrations and bottom-mounted instrument tube penetrations, in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (1995 edition through the 1996 addenda). The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.19 Reactor Vessel Internals

The program includes: 1) augmented VT-1 examinations of reactor vessel internals components by techniques yet to be developed, and 2) monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 to ensure the long-term integrity and safe operation of pressurized water reactor (PWR) vessel internal components. The program includes monitoring industry initiatives to develop enhanced visual examination techniques capable of detecting features on the order of .0005 inches in dimension.

A2.1.20 Reactor Vessel Surveillance

The program provides for periodic testing of metallurgical surveillance samples to monitor the progress of neutron embrittlement of reactor pressure vessel materials as a function of neutron fluence in accordance with Regulatory Guide 1.99, Rev. 2.

A2.1.21 Spent Fuel Pool Neutron Absorber Monitoring

The program monitors long-term performance of borated stainless steel (BSS) panels, credited as a neutron absorber in portions of the spent fuel pool (soluble boron is credited in the rest of the pool). Borated stainless steel surveillance coupons are periodically removed and examined to evaluate coupon thickness and weight loss. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.22 Steam Generator Tube Integrity

The program incorporates the guidance of NEI 97-06 and EPRI TR-107569 for maintaining the integrity of steam generator tubes. The effects of aging are managed by a balance of prevention, inspection, assessment, repair, and leakage monitoring measures. Plant Technical Specifications assure timely assessment of tube integrity and compliance with primary to secondary leakage limits.

A2.1.23 Structures Monitoring Program

The Structures Monitoring Program consists of periodic inspection and monitoring of the condition of structures and structural elements as well as selected non-safety component supports to ensure that aging degradation will be detected and corrected prior to loss of intended function. The program is implemented in accordance with 10 CFR 50.65, NUMARC 93-01, Rev. 2, and Regulatory Guide 1.160, Rev. 2. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.24 Systems Monitoring

The program identifies the evidence of age-related degradation on normally accessible exterior surfaces of piping, components and equipment in systems which are within the scope of license renewal. As part of the implementation of 10 CFR 50.65 (Maintenance Rule), specific guidelines for assessing the material condition of systems, structures, and components during system engineer walkdowns were developed. The effects of aging are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation, such as corrosion, cracking, degradation of coatings, sealants and caulking, deformation, debris and corrosion product buildup. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.25 Thimble Tubes Inspection

The program manages the integrity of the incore neutron monitoring thimble tubes, which serve as a portion of the reactor coolant pressure boundary. The program provides for periodic inspections to detect thimble tube wall thinning due to wear caused by flow induced vibration and preventive maintenance such as flushing, cleaning and replacement. Thimble tube wear is detected at locations associated with geometric discontinuities or area changes along the reactor coolant flow path. The program provides for evaluation of inspection results and appropriate corrective actions.

A2.1.26 Water Chemistry Control

The program mitigates the effects of aging by controlling the internal environment of components in the primary, borated, and secondary water systems. Chemical species known to accelerate corrosion (e.g., chloride, fluoride, and sulfate) are controlled within specified limits. The program implements the guidelines in EPRI TR-105714 for primary water chemistry, and TR-102134 for secondary water chemistry. The program provides for assessment and trending of water chemistry and implements corrective action strategies.

A3.0 EVALUATION OF TIME-LIMITED AGING ANALYSES

As part of a License Renewal Application, 10 CFR 54.21(c) requires that an evaluation of time-limited aging analyses (TLAAs) for the period of extended operation be provided. The following TLAAs have been identified and evaluated to meet this requirement.

A3.1 Reactor Vessel Neutron Embrittlement

The following analyses affected by neutron irradiation caused embrittlement that have been identified as TLAAs:

- Upper shelf energy Appendix A3.1.1
- Pressurized thermal shock Appendix A3.1.2
- RCS pressure-temperature operating limits Appendix A3.1.3

A3.1.1 Upper Shelf Energy

The Charpy upper shelf energy (USE) is associated with the determination of acceptable Reactor Vessel toughness during operation. 10 CFR Part 50 Appendix G requires that the reactor vessel beltline materials must have a USE of no less than 50 ft-lb) throughout the life of the reactor vessel, unless otherwise approved by the NRC. If the USE of a reactor vessel beltline material is predicted to not meet Appendix G requirements, then licensees must submit an analysis that demonstrates an equivalent margin of safety at least three years prior to the time the material is predicted to not meet those requirements.

In the event that the 50 ft-lb requirement cannot be satisfied as stated in 10 CFR 50 Appendix G, or by alternative procedures acceptable to the NRC, the reactor vessel may continue to operate provided requirement IV.A.1. of Appendix G is satisfied. This requirement states that an analysis must conservatively demonstrate, making appropriate allowances for uncertainties, the existence of equivalent margins of safety for continued operation. Procedures for the analysis are provided in NUREG-0744. Acceptance criteria are included in ASME Section XI, Appendix K.

The upper shelf energy of the limiting circumferential beltline weld in the Ginna reactor vessel is expected to decrease below 50 ft-lbs during the period of extended operation. In order to demonstrate equivalent margins of safety for continued operation, a low upper-shelf toughness fracture mechanics analysis has been performed (BAW-2425) to evaluate the limiting circumferential beltline weld (SA-847) for ASME Levels A, B, C, and D Service Loadings. The analysis demonstrates that the limiting beltline weld satisfies the Appendix K requirements for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 54 EFPY.

A3.1.2 Pressurized Thermal Shock

The PTS rule, 10 CFR 50.61 provides screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation may not continue without further plant-specific evaluation. The pressurized thermal shock screening criteria are given in terms of reference temperature RT_{PTS}. The screening criteria are 270°F for plates and axial welds, and 300°F for circumferential welds.

The results of the revised PTS analysis for the limiting material have been reviewed for compliance with 10 CFR 50.61. The methodology used in PTS analysis is based on the projected neutron fluence at the end of the period of extended operation and does not rely on plant-specific surveillance data to calculate ΔRT_{PTS}. Although plant-specific surveillance data could have been used, generic data proved to be more conservative. The analysis associated with PTS has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii) and found to be acceptable.

A3.1.3 Pressure-Temperature Limits

10 CFR Part 50 Appendix G requires that the reactor pressure vessel (RPV) be maintained within established pressure-temperature (P-T) limits including during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure (given the required minimum temperature) is reduced.

Heatup and cooldown limit curves are calculated using the adjusted RT_{NDT} corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT} and adding a margin.

The reactor vessel neutron fluence values corresponding to the end of the period of extended operation and the reactor vessel beltline material properties have been calculated consistent with Regulatory Guide 1.190. The revised fluence values have been used to determine the limiting value of RT_{NDT} using the methods of Regulatory Guide 1.99. The limiting value of RT_{NDT} was used to calculate reactor coolant system (RCS) pressure-temperature (P-T) operating limits that are valid through the end of the period of extended operation. Consistent with NUREG-1800 section 4.2.2.1.3.3, it is not necessary to implement P-T limits to carry the reactor vessel through 60 years at the time of application. The updated limits will be contained in a pressure-temperature limit report (PTLR) or in the Technical Specification (TS) prior to the period of extended operation.

The analysis associated with P-T operating limits has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

A3.2 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)

The potential exists for a loss of fracture toughness due to thermal aging of cast austenitic stainless steel (CASS) components. An evaluation of the susceptibility of CASS components at Ginna Station was made, based on the casting method, molybdenum content, and percent ferrite. It was determined that the CASS RCS elbows were susceptible to a loss of fracture toughness due to thermal aging. A plant-specific flaw tolerance evaluation was conducted, and documented in 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. The evaluation concluded that adequate fracture toughness exists for the RCS loop, including the cast elbows, for the period of extended operation (60 years).

A separate evaluation was made for the reactor coolant pump casings. In WCAP-15873, "A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of R.E. Ginna Nuclear Power Plant for the License Renewal Program," May 2002, it was concluded that the primary loop pump casings are qualified to item (d) of ASME Code Case N-481 for the period of extended operation (60 years).

The evaluation associated with thermal aging embrittlement has been found to be acceptable to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

A3.3 Metal Fatigue

The following issues are considered separately under the TLAA for Metal Fatigue:

- ASME Boiler and Pressure Vessel Code, Section III, Class 1 Appendix A3.3.1
- Reactor Vessel Underclad Cracking Appendix A3.3.2
- ANSI B31.1 Piping Appendix A3.3.3
- Accumulator Check Valves Appendix A3.3.4
- Environmentally Assisted Fatigue Appendix A3.3.5

Fatigue is the gradual deterioration of a material that is subjected to repeated cyclic loads. Components have been designed or evaluated for fatigue according to the requirements of applicable codes.

A3.3.1 ASME Boiler and Pressure Vessel Code, Section III, Class 1

The reactor vessel, pressurizer, steam generators, and reactor coolant pumps have been designed in accordance with the requirements of the ASME Boiler and Pressure Vessel

Code, Section III, Class 1. The reactor vessel internals were designed according to Westinghouse criteria which were later incorporated into ASME Boiler and Pressure Vessel Code. Design codes for the above components are identified in UFSAR Table 5.2-3. The ASME Boiler and Pressure Vessel Code, Section III, Class 1 requires a design analysis to address fatigue and establish limits such that initiation of fatigue cracks is precluded.

Fatigue usage factors for critical locations in the NSSS components were determined using design cycles that were specified in the plant design process. These design cycles were intended to be conservative and bounding for all foreseeable plant operational conditions. The design cycles were subsequently utilized in the design stress reports for various NSSS components satisfying ASME fatigue usage design requirements, and became part of the plant Technical Specifications.

Experience has shown that actual plant operation is often very conservatively represented by these design cycles. The use of actual operating history data allows the quantification of these conservatisms in the existing fatigue analyses. To demonstrate that the Class 1 component fatigue analyses remain valid for the period of extended operation, the design cycles applicable to the Class 1 components were assembled. The actual frequency of occurrence for the design basis cycles was determined and compared to the design cycle set. The severity of the actual plant transients was compared to the severity of the design cycles. This comparison was performed in order to demonstrate that on an event-by-event basis the design cycle profiles envelope actual plant operation. In addition, a review of the applicable administrative and operating procedures was performed to verify the effectiveness of the current design cycle counting program.

This review concluded that the existing design cycles and cycle frequencies are conservative and bounding for the period of extended operation. The analyses associated with verifying the structural integrity of the reactor vessel, reactor vessel internals, pressurizer, steam generators, and reactor coolant pumps have been evaluated and determined to remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i). A confirmatory Fatigue Monitoring Program (described in Appendix B of the LRA) has also been implemented at Ginna Station to provide additional assurance that the fatigue analyses remain valid during the period of extended operation.

A3.3.2 Reactor Vessel Underclad Cracking

Underclad cracking has been reported in the low alloy base metal heat-affected zone (HAZ) beneath the austenitic stainless steel weld overlay that is deposited to protect the ferritic material from corrosion.

A re-evaluation (WCAP 15338) of the generic Westinghouse fracture mechanics evaluation (WCAP 7733) concerning the underclad cracking issue has been performed for 60 years of plant operation. It was concluded that "underclad cracks are of no concern to the structural integrity of the vessel for continued plant operation, even through 60 years of operation." WCAP 15338 is bounding for all Westinghouse plants.

The analysis associated with reactor vessel underclad crack growth has been projected to the end of the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(ii).

A3.3.3 ANSI B31.1 Piping

Design requirements in ANSI B31.1 assume a stress range reduction factor to provide conservatism in the piping design to account for fatigue due to thermal cyclic operation. This reduction factor is 1.0 provided the number of anticipated cycles is limited to 7000 equivalent full temperature cycles. This represents a condition where a piping system would have to be cycled approximately once every 3 days over the extended plant life of 60 years. Considering this limit, a review of the ANSI B31.1 piping within the scope of license renewal was performed in order to identify those systems that operate at elevated temperature and to establish their cyclic operating practices. Under current plant operating practices, piping systems within the scope of license renewal are only occasionally subject to cyclic operation. Typically these systems are subject to continuous steady-state operation and vary operating temperatures only during plant heatup and cooldown, during plant transients, or during periodic testing.

The results of the evaluation for ANSI B31.1 piping systems demonstrated that the number of assumed thermal cycles will not be exceeded in 60 years of plant operation except for the Nuclear Sampling System. For all systems except the Nuclear Sampling System, it has been determined that operation can be projected to the end of the period of license renewal, in accordance with 19 CFR 54.21 (c)(1)(ii). For the sampling system, detailed evaluation of operating cycles will be conducted, and reanalysis, repair, or replacement performed prior to the period of extended operation.

A3.3.4 Accumulator Check Valves

Fatigue of components is recognized as time dependent and therefore the analysis was reviewed for fatigue related to these valves. Fatigue failure is based upon the criteria of the cumulative usage factor (CUF). An analysis was performed on the accumulator check valves at Ginna Station. The analysis concludes that the maximum CUF is 0.74 based on specified load conditions.

Plant transients were reviewed to confirm transient limits and total transient counts to date. The load condition occurrences used in the above analysis bound the transient limits monitored by plant procedures. In accordance with 10 CFR 54.21(c)(1)(i), the existing CUF calculations remain valid because the number of assumed transients would not be exceeded during the period of extended operation.

A3.3.5 Environmentally Assisted Fatigue

Generic Safety Issue (GSI)-190, Fatigue Evaluation of Metal Components for 60 Year Plant Life, identifies a concern of the NRC staff about the potential effects of reactor water environments on reactor coolant system component fatigue life during the period of extended operation. GSI-190, which was closed in December 1999, has concluded that environmental effects have a negligible impact on core damage frequency, and as such, no generic regulatory action is required. However, as part of the closure of GSI-190, the NRC has concluded that licensees who apply for license renewal should address the effects of coolant environment on component fatigue life as part of their aging management programs.

Fatigue-sensitive component locations were evaluated in NUREG/CR-6260 for the older vintage Westinghouse plant. These locations were:

1. Reactor vessel shell and lower head (lower shell at the core support pads)
2. Reactor vessel inlet and outlet nozzles
3. Pressurizer surge line (including hot leg and pressurizer nozzles)
4. Reactor coolant piping charging system nozzle
5. Reactor coolant piping safety injection nozzle
6. Residual Heat Removal system Class 1 piping

Environmental fatigue calculations have been performed for Ginna for those component locations included in NUREG/CR-6260 using the appropriate environmental life correction factor formulae contained in NUREG/CR-6583 for carbon/low alloy steel material, or NUREG/CR-5704 for stainless steel material, as appropriate.

Based on these results, all component locations were determined to be acceptable for the period of extended operation, with the exception of the pressurizer surge line.

A3.3.6 Pressurizer Surge Line

A structural evaluation of the Ginna surge line considering the effects of thermal stratification was performed by Westinghouse in 1991. WCAP-12928 describes the stress and cumulative usage factor analysis performed for the surge line in accordance with NRC Bulletin 88-11. The highest CUF was calculated at the surge line nozzle connection to the RCS hot leg. By accounting for the environmental effects of fatigue, it is recognized that the cumulative fatigue usage factor (CUF) for the pressurizer surge line may exceed the Code allowable value of 1.0 during the period of extended operation. Locations on the surge line have been selected for monitoring using the Fatigue Monitoring Program.

The approach developed for managing the environmental effects of fatigue on the pressurizer surge line will include one or more of the following options:

1. Further refinement of the fatigue analysis to lower the CUF to below 1.0,
2. Repair of the affected locations,
3. Replacement of the affected locations, or
4. Manage the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

Should an inspection program be selected to manage environmentally-assisted fatigue for the surge line during the period of extended operation, inspection details such as scope, qualification, method, and frequency will be provided to the NRC for review and approval prior to entering the period of extended operation.

A3.4 Environmental Qualification of Electric Equipment

10 CFR 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants, requires that selected electrical equipment that is relied upon to remain functional during and following a design basis event be environmentally qualified to perform its intended function. Equipment within the scope of the EQ rule has been identified in accordance with 10 CFR 50.49 paragraph (d) and are listed in the Ginna Station EQ Master List. Only the equipment qualification packages which indicate a qualified life of greater than 40 years were reviewed as a Time-Limited Aging Analysis (TLAA). Equipment qualification packages that indicate a qualified life of less than 40 years are not a TLAA as defined in 10 CFR 54.3 and therefore need not be discussed in the context of license renewal. To establish reasonable assurance that the safety related electrical equipment will perform its safety function when exposed to postulated harsh environmental conditions, licensees are required to develop an environmental qualification program. The program must demonstrate that the safety related electrical equipment required to perform the various safety related functions, identified in 10 CFR 50.49, are qualified to perform as intended. The program must maintain the environmental qualification of the equipment for its installed life. Periodic replacement and/or refurbishment of equipment are performed in order to maintain the qualified life of the device. The qualified life of an equipment type is that period of time the equipment is installed, under normal and abnormal plant operating conditions (thermal and radiation exposure), and still be expected to perform its intended function following a postulated design basis event.

The qualified life of an equipment type is determined using the ambient environmental conditions to which it is exposed for the predicted installation period as well as any internal heat rise and cyclic stresses.

EQ reanalyses have been performed to verify extension of EQ qualification to 60 years for most equipment, and shown to be acceptable per 10 CFR 54.21(c)(1)(i) or (c)(1)(ii). Calculations for the balance of the EQ components is continuing, and will be concluded prior to the period of extended operation.

A3.5 Concrete Containment Tendon Prestress

The Ginna Station containment structure is post-tensioned by 160 vertical tendons. The design for the containment provides for prestressing the concrete in the cylinder walls in the longitudinal direction with a sufficient compressive force to ensure that upon application of the design load combinations there will be no tensile stresses in the concrete due to membrane forces.

The prestressing forces of containment tendons decrease over time due to creep and shrinkage of concrete, and stress relaxation of the prestressing steel wires.

One hundred and thirty seven tendons were retensioned in 1979. The remaining twenty three tendons, which had been retensioned in 1969, were not included in the 1979 retensioning activity. Review of tendon surveillance lift-off data indicates that prestressing forces will remain at acceptable levels through the period of extended operation for all tendons except the group of twenty three which were retensioned in 1969. These tendons will be retensioned prior to the end of the current license period. Technical Specifications require that the results of the surveillance be compared with predicted values to verify that prestressing forces are maintained above the minimum design prestress levels.

Based on this review, the program will adequately manage loss of prestress in containment tendons during the extended period of operation in accordance with 10 CFR 54.21(c)(1)(iii).

3.5.1 Containment Tendon Fatigue

Fatigue tests were conducted on tendon wire materials in 1960 by an independent testing lab. The tests indicated that the tendons were capable of withstanding over 2 million cycles at stress levels between 135 and 158 Ksi. The test results were used to conclude that dynamic loads, considering especially pulsating loads resulting from an earthquake, do not jeopardize buttonhead anchorage.

The tendon fatigue test results indicate that the fatigue limit of the tendon wires exceeds, by many orders of magnitude, the total number of cycles that could accumulate through multiple seismic events over 60 years. The seismic fatigue evaluation remains valid through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

3.5.2 Containment Tendon Bellows Fatigue

The allowable radial and vertical displacements of the containment stainless steel tendon bellows are limited to two cycles per year for the 40-year life of the plant. This limits the total number of allowable bellows displacement cycles to 80. Since the completion of construction, displacements at the tendon bellows have occurred due to pressure testing and temperature changes in the cylindrical shell wall due to seasonal variations and reactor shutdown during refueling outages.

The fatigue usage factor of the tendon bellows has been calculated to be .004 over a 60-year period. Therefore, the structural integrity of the tendon bellows will be maintained through the period of extended operation. Thus, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A3.6 Containment Liner Plate and Penetration Fatigue

The containment liner, liner penetrations and liner steel components of the Ginna Station Containment Structure comply with the ASME Code Section III-1965 for pressure boundary and the AISC Code for structural steel. The containment liner and penetrations, including the equipment and personnel hatch penetrations, were designed as Class B Vessels. The Winter 1965 Addenda of ASME Section III, Subsection B, N-1314(a) requires that the containment vessel satisfy the provisions of Subsection A, N-415.1, "Vessels Not Requiring Analysis for Cyclic Operation," in order that Subsection B rules be applicable.

ASME 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 (6) conditions. An analysis was performed which verifies that each of the six conditions are satisfied for the period of extended operation. This analysis demonstrates that the liner and penetrations comply with the ASME Section III - 1965 Code Rules for fatigue through the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

3.6.1 Containment Liner Anchorage Fatigue

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 corresponds to 100,000 stress cycles.

A total of 100,000 stress cycles corresponds to more than 4 full stress cycles per day for 60 years. Fluctuations of temperature and pressure in containment on a daily basis are not significant enough in magnitude to cause four cycles of design basis stress at the liner anchorage weld each day. Therefore, the original fatigue analysis remains valid for the period of extended operation in accordance with 10 CFR 54.21(c)(1)(i).

A3.7 Containment Liner Stress

The containment liner is carbon steel plate conforming to ASTM A442-60T Grade 60 with a minimum yield of 32,000 psi, and a buckling stress of 16,600 psi at operating conditions. The liner plate thickness is 1/4 in. for the base and 3/8 in. for the cylinder and dome. The liner stresses (meridional directions) were calculated to be 4500 psi compression based upon a prestress force of 0.70 fs. The concrete strain due to creep and shrinkage was established as being 320 x 10^{-6} in/in. This increases the liner stress to 14,100 psi at the end of 40 years.

The creep and shrinkage strain occurring over a 60-year plant life was calculated, and the resulting compressive liner stress due to both time-dependent and non-time dependent loads is determined to be 14,870 psi. This liner stress is less than the liner buckling stress of 16,600 psi and therefore the analysis has been projected to the period of extended operation in accordance with 10 CFR 54.21(c)(1)(ii).

A3.8 Other Time-Limited Aging Analyses

A3.8.1 Crane Load Cycle Limit

The estimated number of load cycles for each crane was compared to the number of design load cycles. The comparison demonstrated that all estimated load cycle combinations were well below the upper design loading cycle limit. In addition, the average percent of the rated load lifted was well below 50% of the limit as set forth in the design criteria. Since the number of operating load cycles for the cranes will be fewer than the design cycles and the average percent of rated load lifted is less than 50% for the design load cycles, the crane load cycle limits will remain valid for the period of extended operation in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A3.9 Exemptions

The requirements of 10 CFR 54.21(c) stipulate that the application for a renewed license should include a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and that are based on time-limited aging analyses, as defined in 10 CFR 54.3. Each active 10 CFR 50.12 exemption has been reviewed to determine whether the exemption is based on a time-limited aging analysis. No plant-specific exemptions granted pursuant to 10 CFR 50.12 and based on a time-limited aging analysis as defined in 10 CFR 54.3 have been identified.

A4.0 TLAA SUPPORTING ACTIVITIES

A4.1 Concrete Containment Tendon Prestress

The prestressing forces generated by the containment wall tendons diminish over time due to stress relaxation of the steel tendon wires and shrinkage and creep of the surrounding concrete. The aging management program developed to monitor the prestressing tendon forces ensures that, by periodic surveillance lift-off tests, the trend lines of the measured prestressing forces meet the requirements of 10 CFR 50.55a(b)(2)(viii)(B). If the trend lines cross the predicted lower limits, corrective action such as retensioning will be taken.

A4.2 Environmental Qualification Program

Equipment environmental qualification has been reviewed and one of the following options was used for those components re-analyzed:

- The original environmental qualification qualified life has been shown to remain valid for the period of extended operation.
- The environmental qualification has been projected to the end of the period of extended operation. Reanalysis addresses attributes of analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions.

Calculations are continuing for the balance of Ginna Station EQ equipment using these methods. These analyses will be complete prior to the period of extended operation.

A4.3 Fatigue Monitoring Program

The program is a confirmatory program that monitors loading cycles due to thermal and pressure transients at selected locations on critical reactor coolant system components. The program provides means for evaluating transients using either a stress based or cycle based methodology. The program provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation.

The effects of the reactor coolant environment on component fatigue life are considered by evaluation of a sample of critical components that include, as a minimum, those components selected in NUREG/CR-6260 using the appropriate environmental fatigue correction factors. Formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.

APPENDIX B

AGING MANAGEMENT ACTIVITIES

APPENDIX B
Contents

APPENDIX B
Tables

B1.0 APPENDIX B - INTRODUCTION

B1.1 Overview

Aging management program descriptions are provided in this appendix for each program credited for managing aging effects based upon the aging management review results provided in Section 3.2 through Section 3.7.

Each of the aging management programs described in this section has been compared against the ten attributes which are described in Section A.1, "Aging Management Review - Generic," Table A.1-1, "Elements of an Aging Management Program for License Renewal," of the NUREG-1800, SRP-LR (Reference 1). A description of the program, relevant operating experience associated with the program, and a conclusion section is provided for each program. The conclusion section serves three purposes:

- it describes conformance to the NUREG-1801 program
- it describes enhancements to certain attributes as considered necessary to be consistent with the NUREG-1801 program, and
- it describes exceptions to certain attributes in the NUREG-1801 program not considered necessary to successfully manage aging concerns.

For aging management programs identified in NUREG-1801 which are not specifically used at Ginna Station, a description of alternative programs which manage the applicable aging effects are described.

For new programs not described in NUREG-1801 a description of the program, an evaluation of the ten attributes and a conclusion regarding the adequacy of the program for managing the effects of aging is provided.

The ten attribute definitions are:

Scope of Program

The specific program is identified. The scope of the program includes the specific structures and components for which the program manages the effects of aging.

Preventive Actions

The activities for prevention and mitigation of age-related degradation are described. For condition or performance monitoring programs that do not rely on preventive actions, this information is not provided.

Parameters Monitored or Inspected

The parameters to be monitored or inspected are identified and linked to the degradation of the structure or component intended function(s).

For a condition monitoring program, the designated parameter monitored or inspected is intended to detect the presence and extent of the identified aging effects.

For a performance monitoring program, a link is established between the degradation of the structure or component intended function(s) and the parameter(s) being monitored.

For prevention and mitigation programs, the parameters monitored are the specific parameters being controlled to prevent or mitigate aging effects.

Detection of Aging Effects

Aging effects are detected before there is a loss of the structure and component intended function(s). The parameters to be monitored or inspected are appropriate to ensure that the structure and component intended function(s) will be adequately maintained for the period of extended operation. Methods or techniques (e.g., visual, volumetric, surface) used for detection of aging effects are discussed. Plant-specific program documentation will include additional information such as frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects.

Monitoring and Trending

Monitoring and trending activities are described, and are designed to provide predictability of the extent of degradation and thus effect timely corrective or mitigative actions.

Acceptance Criteria

The 10 CFR 50 Appendix B (Reference 2) acceptance criteria of the program and its basis is described. The acceptance criteria, against which the need for corrective action is generally evaluated, ensure that the structure and component intended function(s) are maintained under all CLB design conditions during the period of extended operation. Plant-specific program documentation will include the description of methodologies for analyzing results against applicable acceptance criteria. These acceptance criteria will provide for timely corrective action before loss of intended function described in the CLB.

Corrective Actions

This attribute describes 10 CFR 50 Appendix B required actions that will be taken when the acceptance criteria are not met. Timely corrective actions, including root cause determination and prevention of recurrence, are part of this attribute.

Confirmation Process

The confirmation process ensures that preventive actions are adequate and that appropriate corrective actions have been completed and are effective.

Administrative Controls

The administrative controls of the program are described. This includes a 10 CFR 50, Appendix B formal review and approval process.

Operating experience

Plant-specific operating experience, including past corrective actions resulting in program enhancements or additional programs, is reviewed and described as appropriate. This information provides objective evidence to support the conclusion that the effects of aging will be managed adequately so that the structure and component intended function(s) will be maintained during the period of extended operation.

The Ginna Quality Assurance Program implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Appendix A.2 of NUREG-1800 (Reference 1).

The Quality Assurance Program includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and non-safety-related structures, systems, and components that are subject to aging management review. In many cases, existing activities were found adequate for managing aging effects during the period of extended operation.

In some cases, aging management reviews revealed that existing programs should be enhanced to adequately manage the effects of aging. Also, in a few cases, new programs will be developed to provide reasonable assurance that aging effects are adequately managed. Each aging management program presented in this appendix is characterized as either an Aging Management Program or as a Time-Limited Aging Analyses Activity that has been credited by a time-limited aging analysis described in (Section 4.0)

B1.2 Operating Experience

Industry operating experience was incorporated into the License Renewal process through a review of industry documents to identify aging effects and mechanisms that could challenge the intended function of systems and structures within the scope of License Renewal. Review of plant specific operating experience was performed to identify aging effects. This review involved electronic database searches of historical information from Ginna Station as well as other information documented in plant records from as early as early as 1970 to 2002. In addition, discussions with system engineers and long time company employees were conducted for identification of any additional aging concerns.

For those materials and environments identified in NUREG-1801 (Reference 3), no additional aging effects requiring management were identified that were not already identified in the GALL. However, additional materials and environments were identified at Ginna Station that were not identified in the GALL but which had aging effects requiring management. These additional materials and environments and associated aging effects are identified in Section 3. The programs identified for aging management are discussed in this appendix.

B1.3 Aging Management Programs

The following aging management programs are described in the sections listed in this appendix. Site specific programs are indicated. All other programs are fully consistent with or are, with some exceptions, consistent with programs in NUREG-1801.

1. Aboveground Carbon Steel Tanks (Section B2.1.1)
2. ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection (Section B2.1.2)
3. ASME Section XI, Subsections IWE & IWL Inservice Inspection (Section B2.1.3)
4. ASME Section XI, Subsection IWF Inservice Inspection (Section B2.1.4)
5. Bolting Integrity (Section B2.1.5)
6. Boric Acid Corrosion (Section B2.1.6)
7. Buried Piping and Tanks Inspection (Section B2.1.7)
8. Buried Piping and Tanks Surveillance (Section B2.1.8)
9. Closed-Cycle (Component) Cooling Water System Surveillance (Section B2.1.9)
10. Compressed Air Monitoring (Section B2.1.10)
11. Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.11)
12. Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section B2.1.12)
13. Fire Protection (Section B2.1.13)
14. Fire Water System (Section B2.1.14)
15. Flow-Accelerated Corrosion (Section B2.1.15)
16. Fuel Oil Chemistry (Section B2.1.16)
17. Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.17)
18. Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section B2.1.18)

19. Loose Part Monitoring (Section B2.1.19)
20. Neutron Noise Monitoring (Section B2.1.20)
21. One-Time Inspection (Section B2.1.21)
22. Open-Cycle Cooling (Service) Water System (Section B2.1.22)
23. Periodic Surveillance and Preventive Maintenance (Section B2.1.23)
24. Protective Coatings Monitoring and Maintenance Program (Section B2.1.24)
25. Reactor Head Closure Studs (Section B2.1.25)
26. Reactor Vessel Head Penetration Inspection (Section B2.1.26)
27. Reactor Vessel Internals (Section B2.1.27)
28. Reactor Vessel Surveillance (Section B2.1.28)
29. Selective Leaching of Materials (Section B2.1.29)
30. Spent Fuel Pool Neutron Absorber Monitoring (Section B2.1.30)
31. Steam Generator Tube Integrity (Section B2.1.31)
32. Structures Monitoring Program (Section B2.1.32)
33. Systems Monitoring (Section B2.1.33)
34. Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section B2.1.34)
35. Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section B2.1.35)
36. Thimble Tube Inspection (Section B2.1.36)
37. Water Chemistry Control (Section B2.1.37)

B1.4 Time-Limited Aging Analyses Support Activities:

1. Environmental Qualification Program (Section B3.1)
2. Fatigue Monitoring Program (Section B3.2)
3. Concrete Containment Tendon Pre-stress(Section B3.3)

B2.0 AGING MANAGEMENT PROGRAMS

Correlation between NUREG-1801 Generic Aging Lessons Learned (GALL) programs and Ginna programs are in Table B2.0-1. For the Ginna Programs, links to appropriate sections of this appendix are provided.

Table B2.0-1 Correlation Between GALL Programs and Ginna Programs

GALL ID NUMBER	GALL PROGRAM	GINNA PROGRAM
XI.M1	ASME Section XI Inservice Inspection, Subsections IWB, IWC, & IWD	ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection (Section B2.1.2)
XI.M2	Water Chemistry	Water Chemistry Control (Section B2.1.37)
XI.M3	Reactor Head Closure Studs	Reactor Head Closure Studs (Section B2.1.25)
XI.M4	BWR Vessel ID Attachment Welds	Not Applicable, Ginna is a PWR.
XI.M5	BWR Feedwater Nozzle	Not Applicable, Ginna is a PWR.
XI.M6	BWR Control Rod Drive Return Line Nozzle	Not Applicable, Ginna is a PWR.
XI.M7	BWR Stress Corrosion Cracking	Not Applicable, Ginna is a PWR.
XI.M8	BWR Penetrations	Not Applicable, Ginna is a PWR.
XI.M9	BWR Vessel Internals	Not Applicable, Ginna is a PWR.
XI.M10	Boric Acid Corrosion	Boric Acid Corrosion (Section B2.1.6)
XI.M11	Nickel-Alloy Nozzles and Penetrations	Reactor Vessel Head Penetration Inspection (Section B2.1.26)
XI.M12	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section B2.1.34)
XI.M13	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) (Section B2.1.35)
XI.M14	Loose Part Monitoring	Loose Part Monitoring (Section B2.1.19)
XI.M15	Neutron Noise Monitoring	Neutron Noise Monitoring (Section B2.1.20)

GALL ID NUMBER	GALL PROGRAM	GINNA PROGRAM
XI.M16	PWR Vessel Internals	Reactor Vessel Internals (Section B2.1.27)
XI.M17	Flow-Accelerated Corrosion	Flow-Accelerated Corrosion (Section B2.1.15)
XI.M18	Bolting Integrity	Bolting Integrity (Section B2.1.5)
XI.M19	Steam Generator Tube Integrity	Steam Generator Tube Integrity (Section B2.1.31)
XI.M20	Open-Cycle Cooling Water System	Open-Cycle Cooling (Service) Water System (Section B2.1.22)
XI.M21	Closed-Cycle Cooling Water System	Closed-Cycle (Component) Cooling Water System (Section B2.1.9)
XI.M22	Boraflex Monitoring	Spent Fuel Pool Neutron Absorber Monitoring (Section B2.1.30)
XI.M23	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems (Section B2.1.18)
XI.M24	Compressed Air Monitoring	Compressed Air Monitoring (Section B2.1.10)
XI.M25	BWR Reactor Water Cleanup System	Not Applicable, Ginna is a PWR.
XI.M26	Fire Protection	Fire Protection (Section B2.1.13)
XI.M27	Fire Water System	Fire Water System (Section B2.1.14)
XI.M28	Buried Piping and Tanks Surveillance	Buried Piping and Tanks Surveillance (Section B2.1.8)
XI.M29	Aboveground Carbon Steel Tanks	Aboveground Carbon Steel Tanks (Section B2.1.1)
XI.M30	Fuel Oil Chemistry	Fuel Oil Chemistry (Section B2.1.16)
XI.M31	Reactor Vessel Surveillance	Reactor Vessel Surveillance (Section B2.1.28)
XI.M32	One-Time Inspection	One-Time Inspection (Section B2.1.21)
XI.M33	Selective Leaching of Materials	Selective Leaching of Materials (Section B2.1.29)

GALL ID NUMBER	GALL PROGRAM	GINNA PROGRAM
XI.M34	Buried Piping and Tanks Inspection	Buried Piping and Tanks Inspection (Section B2.1.7)
XI.E1	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.11)
XI.E2	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits (Section B2.1.12) Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.11)
XI.E3	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (Section B2.1.17)
XI.S1	ASME Section XI, Subsection IWE	ASME Section XI, Subsections IWE & IWL Inservice Inspection (Section B2.1.3)
XI.S2	ASME Section XI, Subsection IWL	ASME Section XI, Subsections IWE & IWL Inservice Inspection (Section B2.1.3)
XI.S3	ASME Section XI, Subsection IWF	ASME Section XI, Subsection IWF Inservice Inspection (Section B2.1.4)
XI.S4	10 CFR 50, Appendix J	ASME Section XI, Subsections IWE & IWL Inservice Inspection (Section B2.1.3)
XI.S5	Masonry Wall Program	Structures Monitoring Program (Section B2.1.32)
XI.S6	Structures Monitoring Program	Structures Monitoring Program (Section B2.1.32)
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	Structures Monitoring Program (Section B2.1.32)

GALL ID NUMBER	GALL PROGRAM	GINNA PROGRAM
XI.S8	Protective Coating Monitoring and Maintenance	Protective Coatings Monitoring and Maintenance Program (Section B2.1.24)
Chapter X		
X.M1	Metal Fatigue of Reactor Coolant Pressure Boundary	Fatigue Monitoring (Section B3.2)
X.E1	Environmental Qualification (EQ) of Electrical Components	Environmental Qualification Program (Section B3.1)
X.S1	Concrete Containment Tendon Prestress	Concrete Containment Tendon Prestress (Section B3.3)
NA	Plant Specific Program	Thimble Tube Inspection Program (Section B2.1.36)
NA	Plant Specific Program	Periodic Surveillance and Preventive Maintenance (Section B2.1.23)
N/A	Plant Specific Program	Systems Monitoring (Section B2.1.33)

B2.1 Aging Management Activities

B2.1.1 Aboveground Carbon Steel Tanks

Program Description

This program relies on periodic system walkdowns to monitor the condition of aboveground carbon steel storage tanks. These walkdowns include an assessment of the condition of tank surfaces protected by paints or coatings, although the coatings themselves are not credited to perform a preventive function. For storage tanks supported on earthen or concrete foundations, corrosion could occur in inaccessible locations, such as the tank bottom. For such inaccessible surfaces, one-time thickness measurements of the tank bottom performed from inside the tank are used to assess the tank bottom condition.

This program is not specifically used for aging management at Ginna Station. Inspection, testing and surveillance activities described under the scope of this program in NUREG-1801 are performed at Ginna Station by the following programs:

- Systems Monitoring (Section B2.1.33)
- One-Time Inspection (Section B2.1.21)

Operating Experience

In accordance with the guidance provided in Generic Letter 98-04, Ginna Station evaluated plant specifications for use of protective coatings inside containment (including those used on carbon steel tanks such as the accumulators). Historical walkdowns to evaluate areas of flaking and peeling paint have discovered isolated areas of degradation, but the quantity has been insufficient to pose a threat to the pump recirculation capability by plugging the sump screens.

Outside containment, periodic assessments of the outside surfaces of tanks are part of the System Engineer Maintenance Rule (10 CFR 50.65) walkdowns. No significant corrosion of tanks has been documented as a result of these walkdowns.

Ginna Station does not have an operating history of tank bottom thickness measurements, but will perform a one-time inspection of the Reactor Makeup Water Tank prior to the period of extended operation.

Conclusion

Results of a one-time inspection of the Reactor Makeup Water Tank bottom will be evaluated and dispositioned in accordance with the Ginna Station Corrective Action Program. In addition, continued inspection and surveillance activities performed during walkdowns in accordance with the Systems Monitoring Program will provide assurance that age-related degradation of external surfaces of aboveground carbon steel tanks will be adequately managed during the period of extended operation.

B2.1.2 ASME Section XI, Subsections IWB, IWC, & IWD Inservice Inspection

Program Description

The Code of Federal Regulations, 10 CFR 50.55a, imposes the Inservice Inspection (ISI) requirements of ASME Boiler and Pressure Vessel Code (the Code), Section XI, for Class 1, 2, and 3 pressure-retaining components and their integral attachments in light-water cooled power plants. Inspection, repair and replacement of these components are covered in Subsections IWB, IWC, and IWD, respectively, in the 1995 Edition of the Code through 1996 Addenda. The program includes periodic visual, surface, and or volumetric examinations and leakage tests of all Class 1, 2, and 3 pressure-retaining components and their integral attachments including welds, pump casings, valve bodies, and pressure-retaining bolting.

The ASME Section XI Inservice Inspection Program in accordance with Subsections IWB, IWC, and IWD has been shown to be generally effective in managing the effects of aging in Class 1, 2, and 3 components and their integral attachments.

Ginna Station has maintained an Inservice Inspection Program in accordance with 10 CFR 50.55a and Technical Specification requirements. The Fourth Ten-Year Interval of the Ginna Station Inservice Inspection Program began on January 1, 2000 and was developed and prepared to meet the requirements of ASME Section XI, 1995 Edition, with 1996 Addenda.

B2.1.2.1 Operating Experience

A thorough review of industry operating experience relating to inservice inspections has revealed numerous incidents of primary pressure boundary degradation that have been reported through NRC generic communications. These incidents may be grouped in the following categories:

- Boric acid corrosion due to leakage at bolted closures and leakage caused by cracking of primary pressure boundary Alloy 600 components such as reactor vessel head CRDM nozzles;
- Cracking due to SCC in safety injection piping, instrument nozzles in safety injection accumulators, and safety-related stainless steel piping systems containing stagnant or essentially stagnant borated water;
- Crack initiation and growth due to thermal and mechanical loading in high-pressure injection and safety injection lines;
- Degradation of steam generator tubing due to PWSCC, ODSCC, IGA, wastage and pitting; denting and cracking of tubes due to carbon steel support plate corrosion; and pitting and cracking of steam generator shell welds.

Review of plant-specific operating experience revealed the following conditions that were discovered by ISI Program examinations:

- Bolting degradation detected by VT-1 examinations and boric acid leakage by VT-2 leakage exams;
- PWSCC, ODSCC, IGA and denting of Alloy 600 steam generator tubing by eddy current examinations;
- Shallow thermal fatigue cracks in S/G feedwater nozzle-to-pipe weld; and
- Original manufacturing flaw indications in the primary inlet nozzle-to-reactor vessel weld (N2B) and pressurizer lower head-to-shell girth weld. These indications were evaluated by fracture mechanics and determined to be acceptable.

The ISI Program at Ginna Station is continually upgraded to account for industry experience and research and is subject to periodic NRC inspections and self-assessments. The ISI Program has provided an effective means of assuring the pressure integrity of Ginna Station Class 1, 2 and 3 systems.

Conclusion

The ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M1, “ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection” (Reference 3). The continued implementation of the ISI Program provides reasonable assurance that aging effects will be managed such that the intended functions of Class 1, 2 and 3 pressure-retaining components and their integral attachments will be maintained during the license renewal period.

B2.1.3 ASME Section XI, Subsections IWE & IWL Inservice Inspection

Program Description

The Code of Federal Regulations, 10 CFR 50.55a, imposes the Inservice Inspection (ISI) requirements of ASME Boiler and Pressure Vessel Code (the Code), Section XI, Subsection IWE for steel containments (Class MC) and steel liners for concrete containments (Class CC). 10 CFR 50.55a also imposes the examination requirements of ASME Section XI, Subsection IWL for reinforced and prestressed concrete containments (Class CC). The full scope of Subsection IWE includes steel containment shells and their integral attachments; steel liners for concrete containments and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. The scope of Subsection IWL includes reinforced concrete and unbonded post-tensioning systems.

Ginna Station has maintained an Inservice Inspection (ISI) Program in accordance with 10 CFR 50.55a and Technical Specification requirements. The Containment Program which outlines the First IWE and IWL Inservice Inspection Interval requirements for Ginna Station was implemented Sept. 9, 1998 and formally included in the ASME Section XI ISI Program.

The Ginna Station ASME Section XI, Subsection IWE & IWL Inservice Inspection Program (the IWE/IWL Program) manages aging of (a) steel liners of concrete containments and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure retaining bolting, and (b) reinforced concrete containments and unbonded post tensioning systems. The primary inspection methods employed are visual examinations (VT-1, VT-3, VT-1C, VT-3C) with limited supplemental volumetric and surface examinations as necessary. Tendon anchorages and wires are visually examined. Tendon wires are tested for verification that minimum mechanical properties requirements are met. Tendon corrosion protection medium is analyzed for alkalinity content and soluble ion concentrations. Prestressing forces are measured in selected sample tendons.

B2.1.3.1 Operating Experience

A review of industry operating experience relating to degradation of steel containment components and concrete revealed occurrences of corrosion in steel containment shells and liner plates, and degradation of reinforced concrete and pre-stressing systems that have been reported through various NRC generic communications.

Review of plant-specific operating experience revealed the following conditions that were discovered during tendon surveillances and more recent IWE and IWL examinations:

- Loss of pre-stress in most containment tendons requiring re-tensioning of 137 tendons;
- Containment moisture barrier found to be out of conformance with drawing; loose insulation; non-conformance corrected by recaulking;
- Minor corrosion of steel containment liner; wall thickness verified by UT; restoration of protective paint coating;
- Low grease levels in certain tendon grease cans at top of containment; cans refilled;
- Corroded and leaking tendon fill-port piping; all fill ports repaired.

Continued inspections in accordance with the Ginna Station ASME Section XI, Subsections IWE & IWL Inservice Inspection Program provide an effective means for timely detection and correction of any degradation of the containment pressure boundary, concrete and post-tensioning system.

Conclusion

The ASME Section XI, Subsections IWE & IWL Inservice Inspection Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.S1, "ASME Section XI, Subsections IWE," XI.S2, "ASME Section XI, Subsections IWL," and XI.S4, "10 CFR 50, Appendix J" (Reference 3). The continued implementation of the IWE/IWL Program provides reasonable assurance that aging effects will be managed such that the intended functions of the Containment will be maintained throughout the license renewal period.

B2.1.4 ASME Section XI, Subsection IWF Inservice Inspection

Program Description

The Code of Federal Regulations, 10 CFR 50.55a, imposes the Inservice Inspection (ISI) requirements of ASME Boiler and Pressure Vessel Code (the Code), Section XI, for Class 1, 2, 3 and MC piping and components and their associated supports. Inservice inspection of supports for ASME piping and components is addressed in Section XI, Subsection IWF in the 1995 Edition of the Code (with 1996 Addenda). The program includes periodic visual examinations of supports based on a sampling of the total support population. The sample size varies depending on the ASME Class. The largest sample size is specified for the most critical supports; the sample size decreases for the less critical supports (Class 2 and 3). Discovery of deficiencies during regularly scheduled inspections results in an expansion of inspection scope to assure that the full extent of the deficiencies is identified. Degradation that potentially compromises the support function or load capacity is identified for evaluation.

The ASME Section XI, Subsection IWF Inservice Inspection Program (the IWF Program) has been shown to be generally effective in managing the effects of aging in Class 1, 2, and 3 and MC piping and component supports.

Ginna Station has maintained an Inservice Inspection Program in accordance with 10 CFR 50.55a and Technical Specification requirements. The Fourth Ten-Year Interval of the Ginna Station Inservice Inspection Program began on January 1, 2000 and was developed and prepared to meet the requirements of ASME Section XI, 1995 Edition, with 1996 Addenda.

B2.1.4.1 Operating Experience

A review of industry and Ginna-specific operating experience relating to inservice inspection of piping and component supports has revealed incidents of misalignment, improper hot or cold positions on spring supports and constant load supports, arc strikes, weld spatter, and missing, detached, or loosened support items. These conditions have been corrected in accordance with the requirements of Subsection IWF.

The IWF Program at Ginna Station is continually upgraded to account for industry experience and research and is subject to periodic NRC inspections and self-assessments. The IWF sampling inspections have been effective in managing aging effects for Ginna Station ASME Class 1, 2, 3 and MC supports.

Conclusion

The ASME Section XI, Subsection IWF Inservice Inspection Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.S3, "ASME Section XI, Subsections IWF" (Reference 3). The continued implementation of the IWF Program at Ginna Station provides reasonable assurance that aging effects will be managed such that the intended functions of Class 1, 2, 3 and MC supports will be maintained during the license renewal period.

B2.1.5 Bolting Integrity

Program Description

The Ginna Station Bolting Integrity Program is a comprehensive program which addresses aging management requirements for all bolting on mechanical and structural components within the scope of license renewal. The Bolting Integrity Program is based on industry recommendations and EPRI guidelines for materials selection and mechanical properties, installation procedures, joint/gasket designs, lubricants and sealants, torque and closure requirements, and enhanced inspection techniques. The program includes periodic inspection of closure bolting for indications of cracking, loss of material and loss of preload. Consideration is given to guidance in NUREG-1339 and EPRI NP-5769 (with exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for pressure retaining and structural bolting.

The Bolting Integrity Program credits activities performed under the direction of other aging management programs for managing specific aging effects. These programs include the following:

- ASME Section XI, Subsections IWB, IWC, and IWD Inservice Inspection (1995 Edition with 1996 Addenda) (Section B2.1.2)
- ASME Section XI, Subsection IWF Inservice Inspection (Section B2.1.4)
- Periodic Surveillance and Preventive Maintenance (Section B2.1.23)
- Boric Acid Corrosion (Section B2.1.6)
- Systems Monitoring (Section B2.1.33)
- Structures Monitoring Program (Section B2.1.32)

B2.1.5.1 Operating Experience

A thorough review of industry operating experience has revealed numerous incidents of primary pressure boundary degradation. Various NRC generic communications, including information notices, bulletins and generic letters have been issued on bolting degradation. The majority of these incidents have dealt with boric acid corrosion caused by leakage at bolted closures, stress corrosion cracking of high-strength bolts, and cracking due to fatigue. General corrosion of structural bolting located underwater or in humid environments has also been reported.

Review of plant-specific operating experience revealed the following incidents involving bolting degradation:

- Various incidents involving bolting degradation resulting from boric acid leakage at bolted joints detected by VT-1 visual examinations and VT-2 leakage exams;
- Failures of ASTM A 490 high-strength RCP leg-support anchor bolts due to stress corrosion cracking; other factors contributing to the failures were improper heat-treatment and excessive preload during original installation;
- Linear MT indications in machined reduced-section shank of five steam generator manway bolts attributed to incipient fatigue damage due to multiple loadings during tensioning, coincident with tool marks produced during machining.

EPRI reports NP-5769 and TR-104213 document programmatic guidance and recommendations for addressing industry bolting integrity issues. The Bolting Integrity Program has been developed and implemented in accordance with this guidance and therefore provides an effective means of ensuring bolting reliability during the license renewal period.

Conclusion

The Ginna Station Bolting Integrity Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.M18, "Bolting Integrity" (Reference 3). The continued implementation of the Bolting Integrity Program provides reasonable assurance that aging effects will be managed such that the intended function of closure bolting on Class 1, 2 and 3 and other pressure-retaining components, NSSS component supports, and structural bolting will be maintained during the license renewal period.

B2.1.6 Boric Acid Corrosion

Program Description

The current Ginna Station Boric Acid Corrosion Control Program is described in an Administrative Procedure. It is a comprehensive program that was developed and implemented to meet the recommendations of USNRC Generic Letter (GL) 88-05, to monitor the condition of the reactor coolant system (RCS) pressure boundary components for boric acid leakage. The program identifies carbon steel components within the RCS that are susceptible to corrosion from leakage of boric acid. It also provides for periodic visual inspection of adjacent components, structures, and supports for evidence of leakage and corrosion.

This program will be enhanced to account for boric acid corrosion of non-RCS components located in areas where there is the potential for boric acid leakage, including cable connections, cable trays and other susceptible SSCs.

B2.1.6.1 Operating Experience

A review of industry operating experience has revealed numerous incidents of RCS pressure boundary degradation due to corrosion of external surfaces caused by leakage of borated water. These incidents have ranged from leakage at bolted closures causing wastage of carbon and low-alloy steel bolting to leakage at reactor vessel head penetrations resulting in accumulation of boric acid deposits on the head and severe loss of head wall thickness near the penetrations. Recent NRC communications (BL 2002-01) have focused on reactor vessel head degradation from boric acid corrosion.

Review of Ginna-specific operating experience has revealed various incidents involving bolting degradation resulting from boric acid leaks at bolted closures that were detected in their early stages by VT-1 visual examinations and VT-2 leakage examinations. In these cases, the bolting was replaced as a precautionary measure. Recent inspections of the Ginna reactor vessel head and penetrations revealed no evidence of boric acid leakage. Implementation of the Boric Acid Corrosion Control Program at Ginna Station in response to the guidance in NRC GL 88-05 has been demonstrated to effectively manage the effects of boric acid corrosion on the intended function of RCS components.

Conclusion

The Ginna Station Boric Acid Corrosion Control Program will be consistent with NUREG-1801, GALL, Section XI.M10. The program will be enhanced to account for boric acid wastage of non-RCS components, including cable connectors and cable trays as well as other susceptible SSCs. The program provides reasonable assurance that the aging effects due to boric acid corrosion of SSCs within the scope of the program will be managed such that their intended function will be maintained during the license renewal period.

B2.1.7 Buried Piping and Tanks Inspection

Program Description

This aging management program includes preventive measures to mitigate corrosion and periodic inspections to manage the effects of corrosion on the pressure-retaining capacity of buried carbon steel piping and tanks. These preventive measures are in accordance with standard industry practice for maintaining protective coatings. Buried piping and tanks are inspected when they are excavated during maintenance activities and when a pipe is uncovered for any reason. These are considered inspections of opportunity.

This program is not specifically used for aging management at Ginna Station. The inspection activities described under the scope of this program in NUREG-1801 are performed at Ginna Station by the following program:

- One-Time Inspection (Section B2.1.21).

Operating Experience

Portions of several buried piping systems and tanks have been excavated at Ginna Station in connection with maintenance activities. Over the years, several sections of the fire-water yard loops have been inspected and replaced with upgraded materials, including piping, fittings, shut-off valves and hydrants. As far back as 1974, a section of the service water discharge header from the Auxiliary Building was excavated and inspected due to preparations for construction of the Standby Auxiliary Feedwater Building. Several other inspection opportunities arose as a result of corrective maintenance, and others, such as those in 1995, were performed during relocation of underground headers due to construction of the concrete pad for the crane used for replacement of steam generators. Portions of the underground service water header was also uncovered and inspected in 1995. Most recently, in November

2001, a yard hydrant and connecting piping/fittings were replaced due to flow blockage identified by periodic flow testing. In the fall of 2001, the Security Diesel Generator underground fuel oil storage tank was replaced with a new tank as a preventive measure. In these cases, the exterior surface condition of the components inspected or replaced was found to be in good condition.

The Emergency Diesel Generator underground fuel oil storage tanks are inspected and leak-tested periodically in accordance with the Periodic Surveillance and Preventive Maintenance Program. No indication of tank wall degradation has been detected by these inspections. An ultrasonic thickness examination (one time inspection) of these tanks will be performed prior to the end of the current license period to assess the condition of the tank wall.

These inspections have provided an effective means of assuring the structural integrity of buried piping and tanks at Ginna Station.

Conclusion

Continued inspections of opportunity and other one-time inspections will provide reasonable assurance that the effects of aging for buried piping and tanks will be adequately managed during the license renewal period.

B2.1.8 Buried Piping and Tanks Surveillance

Program Description

The purpose of this program is to define surveillance and preventive measures which could be used to mitigate corrosion by protecting the external surface of buried carbon steel piping and tanks. It is recommended by NUREG-1801 to use NACE standards RP-0285-95 and RP-0169-96.

RG&E does not employ these standards, or credit the surveillance and preventive measures referenced in these standards, as aging management programs. Thirty three years of operation, and inspection, testing, and surveillance activities, have demonstrated the effectiveness of current Ginna Station programs in maintaining the intended functions of buried carbon steel piping and tanks. Descriptions of these aging management programs employed currently at Ginna Station, and those to be employed, are provided in the following program descriptions:

- ASME Section XI, Subsections IWB, IWC, IWD, Inservice Inspection Program (Section B2.1.2)
- Water Chemistry Control (Section B2.1.37)

- Open-cycle Cooling (Service) Water System (Section B2.1.22)
- Fire Water System (Section B2.1.14)
- Fuel Oil Chemistry (Section B2.1.16)
- One-time Inspection (Section B2.1.21)
- Buried Piping and Tanks Inspection (Section B2.1.7)
- Structures Monitoring Program (Section B2.1.32)
- Periodic Surveillance/Preventive Maintenance (Section B2.1.23)
- Systems Monitoring (Section B2.1.33)

B2.1.9 Closed-Cycle (Component) Cooling Water System

Program Description

The Closed-Cycle (Component) Cooling Water System Program at Ginna Station applies to the Component Cooling Water (CCW) System.

The program includes (a) preventive measures to minimize corrosion and (b) surveillance testing and inspection to monitor the effects of corrosion on the intended functions of the system. The program relies on maintenance of system corrosion inhibitor concentrations within specified limits of Electric Power Research Institute [EPRI] TR-107396 to minimize corrosion. Surveillance testing and inspections for closed-cycle cooling water system components are performed to evaluate system and component performance. These measures ensure that the CCW system and components serviced by the CCW system are performing their functions acceptably.

B2.1.9.1 Operating Experience

An engineering activity was performed in 1997 to address external corrosion of uninsulated CCW piping, due to intermittent condensation. Prior to insulating the piping, UT readings were taken at selected locations, and no significant wall thinning was noted. Installation of the insulation has prevented this problem from re-occurring.

In 1998, the CCW heat exchangers were retubed after thirty years of service to correct significant wall loss of the tubes on the Service Water side. During the retubing process, an internal inspection was made of the CCW heat exchangers and they were found to be in excellent physical condition.

Conclusion

The Closed-Cycle (Component) Cooling Water System Program provides reasonable assurance that the aging effects will be managed so that the components within the scope of License Renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

There are differences between the parameters monitored at Ginna Station and those recommended by NUREG-1801. For example, EPRI TR-107396 is not referenced in Ginna procedures, and the only parameters monitored are pH, corrosion inhibitor concentrations, and radioactivity. Plant operating experience has not demonstrated the need to monitor the additional parameters in the EPRI report, such as corrosion products, calcium, potassium, or refrigerant chemicals.

Other parameters that are not monitored on each cooler or heat exchanger are the inlet and outlet temperatures and differential pressure; however, temperatures and pressures are monitored at selected locations throughout the system and differential pressure is monitored on the Service Water side of the CCW heat exchangers.

Relative to detection of aging effects, NUREG-1801 suggests the use of corrosion coupons. These are not used at Ginna Station; NDE is used at locations where loss of material may occur. Also, Ginna Station does not perform MIC testing on the chromated water in the CCW system; plant and industry experience indicates that the use of potassium dichromate has inhibited MIC growth.

These differences from the GALL have been evaluated and determined to be minor in terms of assuring proper functionality of system components.

B2.1.10 Compressed Alr Monitoring

Program Description

This program consists of inspection, testing, and monitoring of the instrument and service air systems to ensure that oil, water, rust, dirt, and other contaminants are kept within specified limits. The purpose of such actions is to ensure that license renewal intended functions could be accomplished for the period of extended operation.

This issue was discussed in detail in NRC Generic Letter 88-14. In RG&E's response to that letter, details were provided of how compressed air systems are operated, tested and maintained at Ginna Station. It was also stated that air-operated valves were verified to fail-safe on loss of air, and that therefore the compressed air systems at Ginna Station did not perform a safety function (letter from RG&E to NRC, June 17, 1991).

In accordance with the above discussion and as demonstrated by the results of the scoping process, it has been concluded that the Plant Air Systems are not within the scope of license renewal.

B2.1.11 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

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 (Reference 5) 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. This program, as described, can be thought of as a sampling program. 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. If an unacceptable condition or situation is identified for a cable or connection in the inspection sample, a determination is made as to whether the same condition or situation is applicable to other accessible or inaccessible cables or connections. Cables and connections used in low-level signal applications that are sensitive to reduction in insulation resistance are included in the scope of this program. Technical information and guidance provided in NUREG/CR-5643 (Reference 14), IEEE Std. P1205-2000 (Reference 15), and SAND 96-0344 (Reference 16), and EPRI TR-109619 (Reference 17) are considered.

B2.1.11.1 Scope of Program

This inspection program applies to accessible electrical cables and connections as well as cables used in low-level signal applications that are sensitive to reduction in insulation resistance (e.g., radiation monitoring and nuclear instrumentation) within the scope of license renewal that are installed or stored in the following plant buildings/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.

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.

B2.1.11.2 Preventive Actions

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

B2.1.11.3 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.

B2.1.11.4 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.

B2.1.11.5 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.

B2.1.11.6 Acceptance Criteria

The accessible cables and connections are to be free from unacceptable, visual indications of surface anomalies, which 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.

B2.1.11.7 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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

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.

B2.1.11.8 Confirmation Process

Confirmation of the effectiveness of the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B (Reference 2).

B2.1.11.9 Administrative Controls

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

B2.1.11.10 Operating Experience

Operating experience has shown that adverse localized environments caused by heat or radiation for electrical cables and connections may exist next to or above steam generators, pressurizers and hot process pipes, such as steam and 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.

Conclusion

This Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program will adequately manage the effects of aging so that there is reasonable assurance that these components will perform their intended functions in accordance with the current licensing basis during the period of extended operation. Accessible electrical cables within the scope of this program will be visually inspected at least once every ten years. This is considered an adequate frequency to preclude failure of the conductor insulation since experience has shown that insulation aging degradation is a slow process. The Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Program is a new program that will be consistent with the NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.E1, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements" (Reference 3).

B2.1.12 Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits

Discussion

Rochester Gas and Electric (RG&E) believes that invoking the NUREG-1801 XI.E1, Electrical Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements program to manage the effects of aging in accessible non-EQ cable and connectors provides reasonable assurance that these SC's will perform their intended function during the period of extended operation.

While RG&E agrees in principle that instrument loop calibrations could detect cable degradation as described in NUREG-1801 and the referenced technical reports, RG&E does not agree that crediting instrument loop surveillance testing as part of a new program for managing those aging effects is beneficial.

The basis for this conclusion includes consideration of the following facts:

- Instrument loops are comprised of many discrete components. Each component is subject to unique factors and many components have the potential of introducing calibration variables (e.g. drift) that mimic the symptoms that would be expected from cable or connector insulation degradation. As acknowledged in NUREG-1801, when a loop is found out of calibration troubleshooting is performed and that trouble shooting would eventually include consideration of the instrument cable.
- Plant Operating Experience has shown that most cable or connector failures detected during instrument calibrations are typically due to maintenance activities themselves, not from component aging.
- As instrument loop components are calibrated, repaired or replaced, after adjustment any meaningful trending information that may yield indication of insulation degradation is effectively re-zeroed. The very act of calibrations themselves will, for a time, mask the symptoms of insulation degradation.
- There is no way to know with prescience why an instrument loop may be exhibiting a particular behavior. The only effective way to ensure that a cable or connector is not developing precursors symptomatic of aging effects that may impact the loop function is to periodically monitor the condition of the cable.

The surveillance of instrument loops is an important part of the plant-licensing basis. However, for the reasons described above, surveillance of instrument loops is not considered an effective tool for managing the effects of aging in the passive long-lived portions of an instrument loop. Experience has shown that the aging effects of cable and connector insulation occur very slowly. Therefore, routine maintenance, calibration and repair activities on the active components in an instrument loop initially work to mask indications of cable and cable and connector insulation degradation. Only after the active portions of a loop can no longer be adjusted to compensate for cable and connector degradation would the passive portions of the instrument loop become suspect. Surveillance provides meaningful information, but that information is primarily used to cause changes to the active portions of an instrument loop. The predominate cause of non-event driven degradation in cable and connector insulation is thermal aging. External inspection of cables and connectors and their host environments identifies the possibility of thermal aging long before instrument loop adjustments can no longer compensate for current leakage. Because of this, RG&E considers that the only legitimate way to ensure the continued functioning of the long-lived passive components are those inspection activities performed under the XI.E1 program, Electrical Cables not Subject to 10 CFR 50.49 Environmental Qualification Requirements.

B2.1.13 Fire Protection

Program Description

The Ginna Station Fire Protection Program includes provisions for aging management of fire barriers and fire pumps. The fire barrier inspection program requires visual periodic visual inspection and functional tests of fire barrier penetration seals, fire barrier walls, ceilings, and floors, and periodic visual inspection and functional tests of fire rated doors to ensure their operability is maintained. The program also requires the fire pumps to be periodically tested, with preventive maintenance and inspections performed to ensure their operability. The program also provides for periodic inspection and testing of the relay room halon fire suppression system.

B2.1.13.1 Operating Experience

A review of previous fire barrier inspection results, action reports, and maintenance work requests provides assurance that fire seals, barriers and walls remain intact to perform their intended function. These inspections have effectively identified event-driven degradation such as torn Hemyc wrap, damaged fire seals, and cracked mortar/caulk in walls consistent with use and operation of the facility. No evidence of age-related degradation has been detected.

Periodic inspection, preventive maintenance, and functional testing of fire pumps provided the data and trending necessary to replace the Diesel Fire Pump engine in 1994.

Inspection results and actions taken as a result of the Fire Protection Program continue to provide an effective means of ensuring the structural integrity of systems and components within the program scope during the license renewal period.

Conclusion

The Fire Protection Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M26, "Fire Protection" (Reference 3). The program provides reasonable assurance that the aging effects for fire seals, fire barriers, fire pumps, and the halon system will be managed such that the intended function of the components within the scope of the program will be maintained during the license renewal period.

B2.1.14 Fire Water System

Program Description

The Fire Water System Program is implemented by the Ginna Station Fire Protection Program which includes provisions for aging management of the fire water system and associated components. These components include sprinklers, nozzles, fittings, hydrants, hose stations, standpipes, fire water storage tank, fire booster pump, etc. System and component testing is conducted in accordance with the applicable National Fire Protection Association (NFPA) codes and standards. The fire water system and associated components are normally maintained at required pressure and monitored such that a loss of system pressure is immediately detected and corrective actions initiated. In addition to the testing performed per NFPA codes, portions of the fire water system are subjected to full flow testing. Also, internal portions of the fire water system are visually inspected when disassembled for maintenance. Volumetric NDE inspections using appropriate techniques are performed on sections of the system piping to detect wall loss and fouling. The flow testing and visual and/or volumetric inspections assure that any wall thinning due to corrosion, microbiologically influenced corrosion (MIC), or biofouling are managed such that the system function is maintained.

B2.1.14.1 Operating Experience

Over the life of the plant, portions of the underground yard hydrant system have been replaced with upgraded components based on the results of flow tests and/or inspections. The fire water storage tank has been periodically inspected and preventive maintenance such as recoating the tank internals has been completed to provide additional protection.

Inspection results and actions taken as a result of the Fire Protection Program continue to provide an effective means of ensuring the structural integrity of systems and components within the program scope during the license renewal period.

Conclusion

The Ginna Fire Water System Program will be consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M27, "Fire Water System" (Reference 3). A review of previous fire water system inspection results, action reports and maintenance work requests, provides assurance that the system and associated components remain capable of performing their intended function in accordance with the fire protection program and applicable NFPA standards. This program will be enhanced to provide for replacement or representative sample testing of sprinklers with a service life of 50 years. This replacement/testing activity will be performed at 10 year intervals following the 50 year in-service testing.The program provides reasonable assurance that the aging effects for the fire water system and associated components will be managed such that the intended function of the components within the scope of the program will be maintained during the license renewal period.

B2.1.15 Flow-Accelerated Corrosion

Program Description

The Ginna Station Flow-Accelerated Corrosion Program is implemented by approved site specific procedures. Flow-Accelerated Corrosion (FAC) is one form of erosion/corrosion that can affect piping and pressure vessels with flowing water or wet-steam. The program is a comprehensive program that addresses erosion/corrosion control measures in accordance with Electric Power Research Institute (EPRI) guidelines in the Nuclear Safety Analysis Center (NSAC)-202L, Revision 2. The program includes the performance of: (a) an analysis to determine critical locations using the predictive CHECWORKS computer code, (b) limited baseline inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm the predictions, or repairing or replacing components as necessary.

B2.1.15.1 Operating Experience

Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC IE Bulletin No. 87-01 and NRC Information Notices (INs)81-28, 92-35, and 95-11), in two-phase piping in extraction steam lines (NRC INs 89-53 and 97-84), and in moisture separation reheater and feedwater heater drains (NRC INs 89-53, 91-18, 93-21, and 97-84).

Review of plant-specific operating experience is highlighted as follows:

- In July, 1986, the reactor was manually tripped due to a large steam leak in a moisture separator drain line elbow. The failure of the elbow was due to wall thinning related to FAC and was caused by impingement of steam from a level control valve located in close proximity to the elbow. The elbow was replaced with chrome-moly material and the piping/valve reconfigured to reduce the impingement.
- In March, 1989, erosion/corrosion program inspections found two instances of wall thinning related to FAC in SG carbon steel blowdown piping. The piping was subsequently replaced with chrome-moly piping materials during the 1990 and 1991 refueling outages.
- In June, 1992, a carbon steel pre-separator flash tank located in the extraction steam system experienced a fishmouth rupture. This caused an unisolable steam leak and resulted in a plant shutdown. The wall loss was attributed to erosion/corrosion (FAC) as a result of two-phased water/steam impingement. The impingement was caused by the configuration of the tank internals. The tank and an identical tank in a duplicate line and inlet piping were redesigned and replaced with chrome-moly components with stainless steel internals. The tanks and associated piping were also added to the erosion/corrosion inspection program.
- In July, 1995, erosion/corrosion inspections identified some additional portions of the SG blowdown system, that were previously replaced with chrome-moly piping, were found to have wall thinning from impingement due to FAC at higher rates than expected. The piping was replaced with stainless steel material and reconfigured to alleviate the impingement.

- In October, 1997, erosion/corrosion inspection found some localized thinning due to FAC in a control valve and reducer in the main feedwater system. The components were repaired.
- In March, 1999, visual inspection found wall thinning due to FAC in a moisture separator drain hole shell resulting from impingement of steam/moisture at the drains. The areas were repaired and the drain holes were reconfigured with stainless steel at new locations to eliminate the direct impingement.

The Ginna Station Flow Accelerated Corrosion Program has been revised and improved over the years as inspection data has been collected and trended. Field verification of materials of construction have been performed and used as inputs to CHECWORKS to update predicted rates of wall loss. New locations have been included in the program based on both industry and plant-specific operating experience.

Conclusion

The Ginna Station Flow-Accelerated Corrosion Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M17, "Flow-Accelerated Corrosion" (Reference 3). The program provides reasonable assurance that the aging effects from FAC will be managed such that the intended function of the piping and components within the scope of the program will be maintained during the license renewal period.

B2.1.16 Fuel Oil Chemistry

Program Description

The Fuel Oil Chemistry Program includes (a) surveillance and maintenance procedures conducted in accordance with plant Technical Specifications to mitigate aging effects such as loss of material due to corrosion and fouling buildup on the internal surfaces of fuel oil storage tanks and associated components in systems that contain fuel oil, and (b) measures to verify the effectiveness of the surveillance/ maintenance activities and confirm the absence of aging effects. The program includes (a) surveillance and monitoring procedures for maintaining fuel oil quality by controlling contaminants in accordance with ASTM Standards D975, D1796, D4057 and D4176 and (b) periodic draining, cleaning and visual inspection of internal surfaces of storage tanks. Supplemental wall thickness measurements (i.e., by UT) may be required at locations where contaminants might accumulate, such as tank bottoms.

B2.1.16.1 Operating Experience

The Diesel Fuel Oil Chemistry Program has been implemented at Ginna Station since the commencement of plant operation. The underground storage tanks have been drained and inspected annually until 1993. Since 1993, annual pressure tests have been performed, and internal inspections are performed on a 10-year frequency. No evidence of degradation of the interior surfaces of either storage tank has ever been observed. No evidence of biological activity has ever been observed.

During the spring RFO in 1987, low day tank levels resulting in loss of normal makeup capacity to the EDGs were determined to be due to partially plugged fuel oil transfer pump suction strainers. The strainers were partially plugged with weld flux and a fibrous material. The weld flux was assumed to have come from original construction and the fibrous material from either cleaning rags or filter material. Corrective actions included draining and inspection of both fuel oil storage tanks, with no problems identified. The strainers were repeatedly cleaned until no further accumulation of debris occurred. In addition, new suction strainers were fabricated and installed, cleaning procedures were revised to prohibit exposure to fibrous contaminants, and additional in-line strainers and screen filters were installed.

A routine NRC inspection in 1990 concluded that the Ginna program for procurement, receipt, sampling and inspection of EDG fuel oil is considered to meet the guidelines of RG 1.137 and assures an adequate supply of proper quality fuel oil to the EDGs.

The Fuel Oil Chemistry Program and periodic storage tank inspections have proven effective in managing the effects of aging resulting from fuel oil contamination at Ginna Station.

Conclusion:

The Fuel Oil Chemistry Program at Ginna Station provides reasonable assurance that the effects of aging will continue to be managed such that no loss of intended function will occur during the period of extended operation. There are some differences between the Ginna Station program and the program described in GALL Section XI.M32. Biocides are not added to the fuel and particulate sampling is not conducted in accordance with modified ASTM D2276. However, plant-specific operating experience demonstrates that the existing program has effectively managed the effects of aging due to fuel oil contamination since the commencement of plant operation.

B2.1.17 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

There are no inaccessible medium voltage (2kV - 15 kV) cables (e.g., installed in conduit or direct buried) within the scope of license renewal. Medium voltage cables and connections subject to an aging management review are not installed in environments that lead to the formation of water trees (i.e. not exposed to significant moisture) therefore no aging management program other than that described in GALL Section XI.E1 is required.

B2.1.18 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

Program Description

This program demonstrates that testing and monitoring programs have been implemented to ensure that cranes are capable of sustaining their rated loads.

It should be noted that many components of a crane system perform their intended functions with moving parts or with a change in configuration, or are subject to replacement based on qualified life. These components are screened out of the license renewal aging management process. This program is primarily concerned with structural components that make up the bridge, trolley, rails, stops, and lifting devices.

B2.1.18.1 Operating Experience

Draft NUREG-XXXX, "Technical Assessment, Generic Issue 186: Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants," January 17, 2002, provides a comprehensive assessment of crane issues. There have been numerous crane incidents, some of which resulted in the publication of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants." Most crane failures are caused by human error (not following procedures, improper test) or design issues (poor engineering). Less than 10% of failures were due to improper maintenance, and most of these were due to electrical malfunctions. There is very little history of wear-related or corrosion-related degradation that has impaired the ability of cranes in the industry to perform their intended functions. A reevaluation of crane operations based on Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," 4/11/96, concluded that although there were some inconsistencies between crane operation and the licensing basis at some nuclear power plants, few changes were required by licensees in their operation of cranes (and none related to age-related degradation).

Only one major crane failure occurred at Ginna Station. During plant construction, a portion of the reactor vessel internals weighing 90 tons was dropped about 6 feet. The cause of failure was attributed to a crane brake failure (crane motor overheated and the electromagnetic brake failed). No experience with crane failures due to age-related degradation such as wear or corrosion has occurred.

Conclusion

The Ginna Station inspection of Overhead Heavy Load and Light Load (related to refueling) Handling Systems Program is consistent with NUREG-1801. The inspection and testing program applied to crane assemblies at Ginna Station provides assurance that the intended functions of the cranes will continue to be met during the period of extended operation.

B2.1.19 Loose Part Monitoring

Program Description

The purpose of this program is to rely on inservice monitoring to detect and monitor loose parts in LWR power plants. NUREG-1801 suggests that this program should include measures to monitor and detect metallic loose parts by using transient signals analysis on acoustic data generated due to loose parts impact.

Ginna Station does not employ a loose part monitoring system for the reactor vessel. There is a loose parts monitoring system employed for the steam generators, which is called the digital metal impact monitoring system (DMIMS). However, use of DMIMS is not considered to be an aging management program at Ginna Station - rather, it is a reactive measurement system to detect failed or FME components that have inadvertently entered the steam generators. Aging management programs related to the reactor coolant system and reactor vessel internals are described in:

- ASME Section XI, Subsections IWB, IWC, IWD, Inservice Inspection Program (Section B2.1.2)

- Water Chemistry Control (Section B2.1.37)
- Reactor Vessel Head Penetration Inspection (Section B2.1.26)
- Thermal Aging Embrittlement of CASS (Section B2.1.34)
- Reactor Vessel Internals (Section B2.1.27)
- Bolting Integrity (Section B2.1.5)
- Steam Generator Tube Integrity (Section B2.1.31)
- Reactor Vessel Surveillance (Section B2.1.28)
- One-time Inspection (Section B2.1.21)
- Periodic Surveillance/Preventive Maintenance (Section B2.1.23)
- Systems Monitoring (Section B2.1.33)
- Thimble Tube Inspection (Section B2.1.36)

B2.1.20 Neutron Noise Monitoring

Program Description

The purpose of this program is to rely on monitoring the excess neutron detector signals due to core motion to detect and monitor significant loss of axial preload at the core support barrel's upper support flange.

RG&E does not include this program as one of the Ginna Station aging management programs. Changes in the support structure are managed by the following aging management programs:

- ASME Section XI, Subsections IWB, IWC, IWD, Inservice Inspection Program (Section B2.1.2)
- Reactor Vessel Internals Program (Section B2.1.27)

Thirty three years of successful operation, including two 10-year detailed inservice inspections of the reactor vessel internals, have demonstrated that additional programs, such as neutron noise monitoring programs, are not required to ensure continued functionality of the core support barrel upper support flange.

B2.1.21 One-Time Inspection

Program Description

The Ginna Station One-Time Inspection Program will include measures to verify the effectiveness of an existing aging management program and confirm the absence of an aging effect. The One-Time Inspection Program will address potentially long incubation periods for certain aging effects and provide a means of verifying that an aging effect is either not occurring or is progressing so slowly as to have negligible effect on the intended function of the structure or component. The program elements will include (a) determination of appropriate inspection sample size based on materials of construction, environment, plausible aging effects, and operating experience, (b) identification of inspection locations, (c) selection of examination technique, with acceptance criteria, and (d) evaluation of results to determine the need for additional inspections or other corrective actions. The inspection sample will include locations where the most severe aging effect(s) would be expected to occur. Inspection methods will include visual (or remote visual), surface or volumetric examination, or other established NDE techniques.

For treated-water systems in stagnant or low flow areas, one-time inspections will provide verification of the effectiveness of water chemistry controls in managing effects of aging. For Class 1 piping and associated components less than 4" NPS in diameter that are not inspected by volumetric techniques during inservice inspections, one-time inspections will confirm that crack initiation and growth due to stress corrosion cracking (SCC) or cyclic loading is not occurring. In addition, one-time inspections will be used to determine whether (a) loss of material due to galvanic corrosion at galvanic couples and (b) loss of material due to selective leaching for gray cast iron or brass components represent significant aging effects requiring management in treated water systems.

B2.1.21.1 Scope of Program

The One-Time Inspection Program will be used to determine the acceptability of components that may be susceptible to various aging effects and to verify that unacceptable degradation is not occurring, thereby validating the effectiveness of an existing aging management program or confirming that there is no need to manage age-related degradation for the period of extended operation. The scope of this program includes the following:

- Verification of the effectiveness of the Water Chemistry Control Program for managing the effects of aging in stagnant or low flow portions of piping, or occluded areas of components, exposed to treated water environments;
- Managing cracking due to SCC or cyclic loading due to thermal fatigue in small bore Class 1 piping (< 4 inches NPS) that is directly connected to the reactor coolant system;
- Managing loss of material due to galvanic corrosion on the internal surfaces of piping and components in treated water systems at locations where galvanic couples are present;
- Managing loss of material and/or loss of structural integrity due to selective leaching on the internal surfaces of piping and components made of gray cast iron, bronze, or brass exposed to treated water or raw water environments.

B2.1.21.2 Preventive Actions

The One-Time Inspection Program will perform inspection activities only.

B2.1.21.3 Parameters Monitored/Inspected

The program will monitor parameters directly related to the degradation of a component, such as loss of material due to pitting, crevice, or galvanic corrosion, loss of material or structural integrity due to selective leaching, and cracking due to SCC or cyclic loading. Inspections will be performed in accordance with qualified NDE procedures and include visual, volumetric, and/or surface techniques.

B2.1.21.4 Detection of Aging Effects

The inspection sample size will be a representative sample of the system population, and, to the extent possible, will include bounding or lead components most susceptible to aging based on time in service, severity of operating environment, and operating experience. For small-bore Class 1 piping, other considerations are accessibility and exposure levels. Volumetric examinations will be used for examinations of small-bore piping and associated components since cracking would be expected to originate at the internal surface of the pipe. Other inspections for detection of loss of material due to pitting, crevice and galvanic corrosion will be performed using visual, surface or volumetric methods, as appropriate. In addition, hardness measurements may be performed to evaluate potential loss of structural and/or pressure boundary integrity due to selective leaching. The inspections will be performed in accordance with qualified procedures using qualified personnel, consistent with the ASME Code and 10 CFR 50, Appendix B.

The one-time inspections will be conducted prior to, but near the end of the current operating license period so as to allow sufficient time for mechanisms with long incubation periods to become active. Inspections will be scheduled to minimize the impact on plant operations.

B2.1.21.5 Monitoring and Trending

The One-Time Inspection Program does not provide guidance for monitoring and trending. However follow-up examinations will be required if unacceptable conditions are discovered, requiring expansion of sample size and locations of inspections.

B2.1.21.6 Acceptance Criteria

Any significant indications or relevant conditions of degradation will be evaluated in accordance with the Ginna Station Corrective Action Program. Required expansion of inspection scope and sample size will be determined by engineering evaluation. Criteria for minimum wall thickness will be based on ASME Code design requirements or approved engineering outputs.

B2.1.21.7 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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.21.8 Confirmation Process

Confirmation of the effectiveness of the One Time Inspection Program will be accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.21.9 Administrative Controls

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

B2.1.21.10 Operating Experience

The One-Time Inspection Program is a new program to be implemented before the end of the current operating period. The inspection scope and techniques that will be employed are consistent with industry practice and have proven effective for timely detection of aging effects.

Conclusion

The Ginna Station One-Time Inspection Program will be consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M32, "One-Time Inspection" (Reference 3). Implementation of the One-Time Inspection Program will provide reasonable assurance that effects of aging such as cracking, loss of material, and loss of structural/mechanical integrity will be managed such that no loss of intended function will occur during the period of extended operation.

B2.1.22 Open-Cycle Cooling (Service) Water System

Program Description

At Ginna Station, the open-cycle cooling water system is called the Service Water System. The Service Water System Reliability and Optimization Program (SWSROP) relies on implementation of the recommendations of the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 89-13 to ensure that the effects of aging on the Service Water System will be managed for the extended period of operation. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion and silting in the Service Water System or structures and components serviced by the Service Water System.

B2.1.22.1 Operating Experience

Heat exchangers have experienced erosion/corrosion of end bells, biofouling build-up, and silt accumulation. Zebra mussels have been found and are controlled by the chlorination system, and periodic cleaning of the heat exchanger tubes. Piping systems have experienced corrosion, pitting, MIC, and sedimentation build-up especially in low flow areas and stagnant dead legs off the main flowstream. These are controlled by flushing, the chlorination system, and inspections. Cavitation/erosion of components is monitored by using established NDE methods and components have been repaired/replaced as necessary.

Ginna Station has also replaced and/or retubed many of these heat exchangers, most notably the Reactor Containment Fan Coolers (replaced), Emergency Diesel Lubricating Oil and Jacket Water heat exchangers (retubed), and the Component Cooling Water heat exchangers (retubed).

The guidance of NRC GL 89-13 has been implemented for approximately 10 years and has been effective in managing and monitoring the effects of aging due to biofouling, corrosion, erosion, and silting in components serviced by the Service Water System.

Conclusion

The Ginna "Service Water System Reliability and Optimization Program" implements the NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M20, "Open-Cycle Cooling Water System" (Reference 3). The SWSROP provides reasonable assurance that aging effects of the Service Water System will be managed so that the components subject to aging management review will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

The program is considered consistent with the GALL, with the following minor differences: 1) heat transfer tests are not performed on selected small heat exchangers which are periodically cleaned and inspected in accordance with the Periodic Surveillance and Preventive Maintenance Program, and 2) the SWSROP does not address protective coatings, which are not credited for aging management in the Ginna Service Water System.

B2.1.23 Periodic Surveillance and Preventive Maintenance

Program Description

The Periodic Surveillance and Preventive Maintenance Program has been a vital element of maintaining plant equipment and structure condition and ensuring their reliability at Ginna Station. The Periodic Surveillance and Preventive Maintenance Program is credited for managing aging effects such as loss of material, crack initiation, fouling buildup, and loss of seal for systems, structures, and components within the scope of license renewal. The program provides for visual inspection and examination of surfaces of selected equipment items and components, including fasteners, for evidence of defects and age-related degradation such as corrosion, wear, cracking, fouling, etc., on a specified frequency based on operational experience. Leak inspection of piping and components in selected portions of systems are also performed on a specified frequency. Additionally, the program provides for replacement or refurbishment of certain components on a specified frequency based on operational experience. The Periodic Surveillance and Preventive Maintenance Program is also used to verify the effectiveness of other aging management programs.

B2.1.23.1 Scope of Program

The Periodic Surveillance and Preventive Maintenance Program provides for visual inspections and surface examinations of certain piping, equipment and components in all plant systems within the scope of license renewal. Additionally, the Periodic Surveillance and Preventive Maintenance Program provides for replacement or refurbishment of certain components on a specified frequency.

B2.1.23.2 Preventive Actions

The inspection and testing activities required by the Periodic Surveillance and Preventive Maintenance Program are primarily monitoring activities. However, periodic replacement or refurbishment of components may be considered preventive in nature.

B2.1.23.3 Parameters Monitored/Inspected

The administrative procedures that govern the Periodic Surveillance and Preventive Maintenance Program provide instructions for monitoring systems, structures and components to permit early detection of degradation. The program provides for visual inspection and examination of surfaces of selected equipment items and components, including fasteners, for evidence of defects and age-related degradation such as loss of material due to corrosion and wear, cracking, fouling buildup, and leakage, on a specified frequency based on operational experience. Equipment or systems operating parameters, e.g., pressure, flow, and temperature, are monitored by performance tests. These tests are effective in detecting performance degradation that may be indicative of aging effects.

Current guidelines in operations, maintenance, and surveillance test procedures and plant work orders will be enhanced to provide explicit guidance on detection of applicable aging effects and assessment of degradation.

B2.1.23.4 Detection of Aging Effects

Aging effects such as loss of material due to corrosion and wear, cracking, loss of seal, etc., are detected by visual inspection of surfaces for evidence of leakage, material thinning, accumulation of corrosion products, and debris, Operations, maintenance, and surveillance test procedures and task descriptions will be enhanced to provide explicit guidance on detection of applicable aging effects and assessment of degradation.

Administrative procedures that govern the Periodic Surveillance and Preventive Maintenance Program provide for evaluation of frequency and appropriateness of Periodic Surveillance and Preventive Maintenance activities to assess effectiveness and compare with typical industry practices.

B2.1.23.5 Monitoring and Trending

The Periodic Surveillance and Preventive Maintenance Program provides for monitoring and trending of material condition and equipment performance. PSPM activity intervals are established to provide timely detection of degradation and are based on service environment as well as industry and plant-specific operating experience and manufacturers recommendations. Operations and maintenance procedures specify activities such as periodic plant walkdowns for monitoring systems, structures and components for early detection of degradation such as coatings failures, corrosion, cracking, leakage and physical condition, mechanical damage, and loose or missing hardware. Data from walkdowns are documented, trended and evaluated to identify and correct deficiencies. Periodic Surveillance and Preventive Maintenance intervals may be adjusted as necessary based on inspection results and industry experience.

B2.1.23.6 Acceptance Criteria

Operations, maintenance, and surveillance procedures and specific task instructions will be enhanced to include explicit instructions for detection of aging effects and definition of acceptance criteria. Degradations deemed to be unacceptable are addressed by the ACTION Reporting process under the Corrective Action Program.

B2.1.23.7 Corrective Actions

Any identified condition that is determined to be deficient or unacceptable is addressed and evaluated under the Corrective Action Program. 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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.23.8 Confirmation Process

Confirmation of the effectiveness of the Periodic Surveillance and Preventive Maintenance Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B. The ACTION Report disposition process includes checks, follow-up inspections, and reviews to verify the adequacy of proposed corrective actions.

B2.1.23.9 Administrative Controls

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

B2.1.23.10 Operating Experience

Periodic Surveillance and Preventive Maintenance activities have been in place since Ginna Station began operation. Review of plant-specific operating experience reveals that significant numbers of ACTION Reports and Plant Work Orders have been generated to correct conditions identified as a result of

Periodic Surveillance and Preventive Maintenance Program activities. These activities have proven to be effective in maintaining the material condition of systems, structures and components and detecting unsatisfactory or degraded conditions.

Conclusion

The Ginna Station Periodic Surveillance and Preventive Maintenance Program, with enhancements identified above, has been evaluated using the generic program attributes identified in Appendix A of the SRP. The continued implementation of the Periodic Surveillance and Preventive Maintenance Program provides reasonable assurance that aging effects will be managed such that the intended function of systems and components within the scope of license renewal will be maintained during the extended period of operation.

B2.1.24 Protective Coatings Monitoring and Maintenance Program

Program Description

Proper maintenance of protective coatings inside containment (described as Service Level 1 in NRC Regulatory Guide 1.54, Rev. 1) is essential to ensure operability of post accident safety systems that rely on water recirculated through the containment emergency Sump "B." Ginna Station maintains protective coatings inside containment in accordance with our program as described in our December 1, 1998 response to Generic Letter 98-04, to ensure that paint chips or flakes do not dislodge in a post-accident environment and cause unacceptable sump blockage.

This program is not considered a license renewal aging management program. However, to demonstrate compliance with the resolution of GSI-191, the 10 element attributes of NUREG-1801 are discussed below:

B2.1.24.1 Scope of Program

All coatings inside containment that are procured, applied, and maintained by Ginna Station or our contractors are within the scope of the program. In addition, on a periodic basis consistent with refueling outages, a comprehensive containment walkdown is performed to look for all loose or flaking protective coatings, to minimize the potential for clogging of the containment sump "B" screens.

B2.1.24.2 Preventive Actions

This program is preventive in nature. All loose protective coatings are removed from the containment prior to startup from each refueling outage. Flaking/peeling paint areas are evaluated, and repaired or replaced if it is considered to be necessary.

B2.1.24.3 Parameters Monitored/Inspected

For protective coatings, degradation is considered to be visible defects, such as blistering, cracking, flaking, peeling, rusting, and physical damage.

B2.1.24.4 Detection of Aging Effects

Ginna Station periodically conducts visual inspections inside containment. General conditions, including coating conditions, are observed during the VT-2 leakage examination of Class 1 components and piping prior to startup after each refueling outage and during the VT-2 leakage examination of Class 2 and 3 piping, supports, and attachments. General walkdowns by Operations, Performance Monitoring, Systems Engineering, Radiation Protection, and Maintenance personnel, as well as crane inspections prior to refueling outages, ensure a general awareness of conditions by a variety of observers. If a localized area of degraded coating is identified, that area is evaluated and scheduled for repair or replacement, as necessary. These observations, and the resulting repair/replacement activities, assure that the amount of Service Level 1 coatings which may be susceptible to detachment from the substrate during a LOCA event is minimized. Ginna Station has incorporated standard industry guidance on coatings in site-specific procedures.

Ginna Station has also completed the baseline inspection required by ASME Section XI, Subsections IWE and IWL. These inspections provided the parameters to be evaluated for (loose/missing parts, corrosion, erosion, etc.) on the containment steel and concrete surfaces.

B2.1.24.5 Monitoring and Trending

The routine inspections performed and evaluated as described in Paragraph B2.1.25.5 above are compared to previous inspections from earlier refueling stages. The need for repair/replacement of degraded protective coating is evaluated, and scheduled as considered necessary.

For IWE/IWL inspections, where no degradation or defects were identified, a five-year inspection interval is used.

B2.1.24.6 Acceptance Criteria

Visual observations of the condition of protective coatings are made by qualified Systems and Structural Engineers, performed in accordance with checklists provided in site-specific procedures for structural monitoring and system engineering walkdowns. Major degradation is documented on an ACTION Report (see B2.1.23.8 below), and repaired as required.

B2.1.24.7 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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.24.8 Confirmation Process

Confirmation of the effectiveness of the Protective Coatings Monitoring and Maintenance Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.24.9 Administrative Controls

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

B2.1.24.10 Operating Experience

Although Ginna Station was designed and built prior to Regulatory Guide 1.54 and related ANSI Standard N101.4, great care was taken in the selection and application of protective coatings within containment.

In the construction of Ginna Station, contemporary standards were specified to ensure that protective coatings applied would perform their functions under environmental conditions experienced during operation and the design-basis accident and to do so without hazard of interfering with other nuclear components.

Construction painting standards included techniques for preparation of surfaces to be painted, sampling, thickness measurement and control, and a detailed paint schedule including components and paint materials for plant structures and equipment. In addition, separate specifications for the preparation, application, material, and paint sampling for the interior of the containment dome were followed.

The painting of the containment structure and components inside the containment was governed by a Westinghouse process specification. This specification covered the application of paint systems to equipment and structures in containments which use additive spray systems for fission product removal and/or containment cooling.

For Ginna Station, Service Level 1 coatings are subject to the requirements of site specific procedures. Adequate assurance that the applicable requirements for the procurement, application, inspection, and maintenance of protective coatings are implemented is provided by procedures and programmatic controls, approved under the R. E. Ginna Nuclear Power Plant Quality Assurance program.

Service Level 1 coatings used for new applications or repair/replacement activities are procured from a vendor with a quality assurance program meeting the applicable requirements of 10 CFR Part 50 Appendix B. The applicable technical and quality requirements that the vendor is required to meet are specified by Ginna Station. Acceptance activities are conducted in accordance with procedures that are consistent with ANSI N 45.2 requirements (e.g., receipt inspection, source surveillance, etc.). This specification of required technical and quality requirements combined with appropriate acceptance activities provides adequate assurance that the coatings received meet the requirements of the procurement documents.

The qualification testing of Service Level 1 coatings used for new applications or repair/replacement activities inside containment meets the applicable requirements contained in the standards and regulatory commitments referenced. Service Level 1 coatings for Ginna Station are procured from the Carboline Company. The Carboline Company was last assessed by a Quality Control Procurement Audit in Spring, 1999.

The surface preparation, application and surveillance during installation of Service Level 1 coatings used for new applications or repair/replacement activities inside containment also meet the applicable portions of the standards and regulatory commitments referenced. Documentation of completion of these activities is performed consistent with the applicable requirements.

The investigation of materials compatibility in the post-accident design-basis environment included an evaluation of protective coatings for use in the containment. The results of the protective coatings evaluation showed that several inorganic zincs, modified phenolics, and epoxy coatings are resistant to an environment of high temperature (320°F maximum test temperature) and alkaline sodium borate. Long-term tests included exposure to spray solution at 150°F to 175°F for 60 days, after initially being subjected to the design-basis accident cycle. The protective coatings, which were found resistant to the test conditions (that is, exhibited no significant loss of adhesion to the substrate nor formation of decomposition products), comprise virtually all of the protective coatings used in the Ginna containment. Hence, the protective coatings will not add deleterious products to the core cooling solution. Essentially all carbon steel surfaces are coated with Carbozinc-11 (inorganic zinc primer) and Phenoline 305 (modified phenolic top coat). Phenoline 305 protective coating is also used on concrete surfaces.

Several test panels of the types of protective coatings used at Ginna Station were exposed for two design-basis accident cycles and showed no deterioration or loss of adhesion with the substrate.

In the safety evaluation of the SEP Topic VI-1, Organic Materials and Post-Accident Chemistry dated February 19, 1982, plant design was reviewed with respect to the effect of paints and coatings under accident conditions. Phenolic based paints are among the most radiation resistant, remaining serviceable after radiation dosage in excess of 10^9 rad. For a severe Design Basis Accident (DBA), 10^8 rad would be a conservative dose estimate. Most painted areas are calculated to receive less than 10^7 rad.

On the basis of the above information, the NRC found, in SEP Topic VI-1, that there is reasonable assurance that the radiation, thermal, and chemical resistance of the organic coatings used in the plant is sufficiently high that deterioration under DBA conditions would not interfere with the operation of engineered safety features. Qualification tests demonstrated that the types of organic coating materials used in the containment will maintain their integrity and remain in serviceable conditions after exposure to the severe environmental conditions of a DBA.

Conclusion

The Protective Coatings Monitoring and Maintenance Program inside containment, although not developed in accordance with Regulatory Guide 1.54 and ASTM D5163-96, is consistent with the NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.S8, "Protective Coatings Monitoring and Maintenance" (Reference 3). Our program was described in detail in our December 1, 1998 response to Generic Letter 98-04, and was accepted by the NRC in their letter of November 19, 1999. Although consistent with NUREG-1801 it is not considered an aging management program, but is described to demonstrate compliance with the resolution of GSI-191.

B2.1.25 Reactor Head Closure Studs

Program Description

This program includes (a) inservice inspection (ISI) in accordance with the requirements of the ASME Code, Section XI, Subsection IWB (1995 edition through the 1996 addenda), Table IWB2500-1, and (b) preventive measures to mitigate cracking.

The ISI portion of the program is described in its entirety in the program description for "ASME Section XI, Subsection IWB, IWC, IWD, Inservice Inspection". (Section B2.1.2)

The reactor head closure studs are fabricated from ASME SA-320 Grade L43 (AISI 4340) low-alloy steel and thus are not susceptible to stress corrosion cracking (specified minimum yield strength of 105 ksi). A comprehensive discussion of this subject is provided in the program description for "Bolting Integrity" (Section B2.1.5).

B2.1.26 Reactor Vessel Head Penetration Inspection

Program Description

The program includes performing (a) primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components, (b) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and (c) inservice inspection (ISI) of reactor vessel head penetrations and bottom-mounted instrument tube penetrations, in accordance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (1995 edition through the 1996 addenda) to detect PWSCC and its effect on the intended function of the component.

Primary water chemistry is monitored and maintained in accordance with the Water Chemistry Control Program, described in (Section B2.1.37) of this appendix.

B2.1.26.1 Scope of Program

The program is focused on managing the effects of crack initiation and growth due to PWSCC of the reactor vessel head and bottom-mounted instrumentation penetrations of the Ginna reactor vessel. In response to the industry-wide initiative relative to GL 97-01, "Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations," a comprehensive eddy current inspection of all the Alloy 600 vessel closure head penetrations was performed at Ginna Station in 1999. The bottom-mounted instrumentation penetrations are routinely examined in accordance with ASME Section XI, Subsection IWB-2500-1. Since these penetrations operate at a lower RCS temperature, there has been no significant degradation of these penetrations in the industry.

B2.1.26.2 Preventive Actions

Preventive measures to mitigate PWSCC are in accordance with the Water Chemistry Control Program. The eddy current inspections performed in 1999 were not preventive, but provided excellent indication that no cracking had occurred in these nozzles.

As a result of these examinations, and industry-wide concern as expressed in NRC Bulletins 2001-01 and 2002-01, Ginna Station has decided to replace the reactor vessel head and CRDM penetrations. This replacement is scheduled for the fall 2003 refueling outage.

B2.1.26.3 Parameters Monitored/Inspected

The purpose of the program is to detect PWSCC so as to control or repair degradation which could have a negative effect on the intended pressure boundary function of the reactor vessel head penetrations.

B2.1.26.4 Detection of Aging Effects

The purpose of the eddy current examinations performed in 1999 was to detect PWSCC-initiated cracks in the CRDM penetrations. No significant indications were discovered during these inspections.

The reactor vessel head is planned to be replaced in 2003, with Alloy 690TT material used for all penetrations. When installed, Ginna Station will continue to follow industry events and developments and reevaluate the type, need, and schedule for addition inspections.

B2.1.26.5 Monitoring and Trending

The 1999 volumetric examination indicated no significant degradation of the reactor vessel head penetrations. The next step in this process is to replace the reactor vessel head and CRDMs in 2003. Ginna Station will follow industry events and developments to determine further inspection activities.

B2.1.26.6 Acceptance Criteria

The results of the 1999 volumetric examinations were analyzed, and it was determined that there was no significant degradation of the reactor vessel head penetrations. Safe operation of the station could continue until the planned vessel head replacement in 2003.

B2.1.26.7 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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.26.8 Confirmation Process

Confirmation of the effectiveness of the Reactor Vessel Head Penetration Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.26.9 Administrative Controls

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

B2.1.26.10 Operating Experience

Significant operating experience has been documented for PWSCC of Alloy 600 vessel head penetrations, particularly GL 97-01, and Bulletins 2001-01 "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," 8/3/01 and 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, 3/18/02." Comprehensive eddy current examinations were performed at Ginna Station in 1999 as an industry initiative in response to GL 97-01. As noted earlier, no significant degradation was evidenced. However, in response to industry concerns in this area. Ginna Station has proactively planned to replace the reactor vessel head, with penetrations using Alloy 690TT material. This replacement is scheduled for fall 2003.

Conclusion

The Ginna Station ASME Section XI ISI program has been effective in maintaining the intended function of the current reactor vessel upper and lower head penetrations. The type and extent of inspections to be performed under the Reactor Vessel Head Penetration Inspection Program for the new reactor vessel head will be determined as Ginna Station continues to follow industry events and developments.

B2.1.27 Reactor Vessel Internals

Program Description

The Ginna Station Reactor Vessel Internals Program includes (a) monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 to ensure the long-term integrity and safe operation of PWR reactor vessel internal (RVI) components, and will include (b) augmentation of the ASME Section XI, Subsection IWB, Table IWB-2500-1 Inservice Inspection (ISI) Program (1995 Edition with 1996 Addenda) for certain susceptible or limiting components or locations,. Detection of fine cracks in non-bolted components will be achieved by augmenting the ASME Section XI ISI Program with enhanced visual methods capable of resolving .0005 inch features of interest when cost effective techniques become available. Inspection and replacement of baffle-former bolts was performed in 1999, and the results are considered acceptable. There are no future plans for inspection/replacement of baffle-former bolts at Ginna Station. However, ongoing industry initiatives will be monitored and the Reactor Vessel Internals Program will be modified appropriately to incorporate industry lessons learned.

B2.1.27.1 Scope of Program

This program will be focused on managing the effects of crack initiation and growth due to stress corrosion cracking (SCC) and irradiation assisted stress corrosion cracking (IASCC), and loss of fracture toughness due to neutron irradiation embrittlement or void swelling. The program includes preventive measures for mitigation of SCC and IASCC. In addition, the program will include augmented ISI inspections to monitor the effects of cracking on internals components. Repair and/or replacement activities may be required to maintain RVI intended function. Since cracks would be expected to initiate at surfaces, augmented visual examinations would provide the required detection capability.

The program will include the following elements: (a) identification of the most susceptible or limiting RVI components and locations; (b) development of appropriate inspection techniques to permit detection and characterizing features of interest (fine cracks) and demonstration of effectiveness of the proposed techniques; and (c) implementation of the augmented inspections during the license renewal term.

The following RVI components are judged to be most susceptible to crack initiation and growth due to IASCC and loss of fracture toughness due to neutron irradiation embrittlement and/or void swelling:

- Lower core plate and fuel alignment pins;
- Lower support columns;
- Core barrel and core barrel flange in active core region;
- Baffle and former plates;
- Thermal shield and neutron panels;
- Bolting - lower support column, baffle-former, and barrel former

It is noted that loss of fracture toughness/cracking of cast austenitic stainless steel (CASS) RVI components due to synergistic effects of neutron embrittlement and thermal aging is not an aging effect requiring management for the Ginna Station internals since there are no RVI components within the scope of license renewal which are fabricated from CASS.

B2.1.27.2 Preventive Actions

The RVI program will be primarily an inspection program. However, monitoring and control of reactor coolant water chemistry parameters in accordance with EPRI TR-105714 is a preventive measure which reduces susceptibility of RVI components to crack initiation and growth due to SCC and IASCC.

B2.1.27.3 Parameters Monitored/Inspected

The program will monitor the effects of cracking on RVI components by detection and sizing of cracks using augmented ISI techniques such as enhanced visual and ultrasonic methods according to the requirements of ASME Section XI, Table IWB-2500-1.

B2.1.27.4 Detection of Aging Effects

Reactor vessel internal components are inspected in accordance with the inservice inspection requirements of ASME Section XI, Subsection IWB, Examination Category B-N-3 for all accessible surfaces of reactor core support structures that can be removed from the vessel. ASME Section XI specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (a) verifying parameters such as clearances, settings and physical displacements, and (b) detecting discontinuities and imperfections such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear or erosion.

Augmented examinations such as enhanced visual techniques will be required for detection of fine or tight cracks. The enhanced visual technique must be capable of .0005 inch resolution in order to detect flaws sufficiently small in size to preserve intended function. Detection of cracking in bolts and fasteners is generally not possible by visual techniques because cracking typically occurs at the bolt head/shank intersection, which is not accessible for visual examination. Ultrasonic examination techniques are required for detecting cracks in bolts and fasteners used to secure bolted connections such as baffle/former bolts and other bolts.

The VT-3 examination per ASME Section XI, Subsection IWB, Category B-N-3 is performed once per 10-year interval on each part of the RVI. Augmented examinations for detection of cracking in components susceptible to IASCC or loss of fracture toughness due to irradiation embrittlement will be scheduled as either periodic or one-time inspections. With respect to augmented examinations of component types susceptible to IASCC and irradiation embrittlement, components with the highest susceptibility (as determined by fluence, temperature and stress) will be selected for examination. If these leading components are found to be free of cracking, less susceptible components may not require examination. The scheduling of future augmented examinations will depend on the results of the initial examinations.

B2.1.27.5 Monitoring and Trending

The Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Issue Task Group (ITG) on Reactor Vessel Internals is currently sponsoring research with the aim of acquiring more data on IASCC, SCC, neutron embrittlement, synergistic effects of neutron embrittlement plus thermal aging, void swelling, and stress relaxation of PWR RVI. Much of this data will be gained through analysis of material currently being irradiated in both research reactors and commercial reactors worldwide. The results of this data will assist Ginna Station in determining the scope and schedule of augmented examinations for cracking, change in dimensions, and loss of preload due to stress relaxation. Ginna Station will actively monitor the EPRI MRP ITG and Westinghouse Owners Group work so as to gain the full benefit of the data that will be generated.

B2.1.27.6 Acceptance Criteria

Any indication or relevant condition of degradation will be evaluated in accordance with IWB-3100, which refers to acceptance standards contained in IWB-3400 and IWB-3500.

B2.1.27.7 Corrective Actions

Repair and replacement activities are performed as required by ASME Section XI. Repairs are conducted in accordance with the requirements of IWB-4000 and replacements in accordance with IWB-7000.

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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.27.8 Confirmation Process

Confirmation of the effectiveness of the Reactor Vessel Internals Program will be accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.27.9 Administrative Controls

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

B2.1.27.10 Operating Experience

A thorough review of industry operating experience was performed, including NRC generic communications and other industry event reports. Two significant NRC communications reviewed were NRC Information Notice 98-11, "Cracking Of Reactor Vessel Internal Baffle Former Bolts In Foreign Plants," and NRC Information Notice 84-18, "Stress Corrosion Cracking in PWR Systems." Most of the industry operating experience reviewed has involved cracking of austenitic stainless steel baffle-former bolts, reportedly due to IASCC, and SCC of high-strength internals bolting. SCC of guide tube split pins fabricated from Alloy X-750 has also been reported. Split pin failures were attributed to improper heat treatment and excessive installation preloads, resulting in increased susceptibility to SCC.

A review of Ginna Station plant-specific experience with reactor vessel internals indicates that Ginna has responded proactively to industry experience with respect to reactor internals degradation. Two examples of this proactive response are as follows:

- Guide-tube split pins were preemptively replaced at Ginna during the 1986 refueling outage using a pin of improved design and heat treatment; however, there is no evidence that split pin failures actually occurred at Ginna.
- Augmented examination and preemptive replacement of selected baffle-former bolts was performed at Ginna in 1999. Out of a total population of 728 Type 347 stainless steel bolts, 639 were examined by UT. Approximately 9% of these bolts exhibited defect-like indications. A total of 56 Type 316 stainless steel replacement bolts were installed. These were bolts that contained defect-like indications and were part of a pre-qualified minimum bolt pattern for two-loop nuclear plants that was generated by the Westinghouse Owners Group (WCAP-15036). Maintaining the structural integrity of the bolts within this pattern assured compliance with requirements of ASME III, Subsection NG (1989).

Results of ongoing research being conducted by Electric Power Research Institute, Reactor Vessel Internals Issue Task Group (EPRI RI-ITG) on aging effects of reactor vessel internals will be closely followed to assure that guidance on corrective action for these aging effects is incorporated into aging management program activities at Ginna Station. Ginna Station also participates in Westinghouse Owner's Group activities related to reactor vessel internals.

Conclusion

The Ginna Station Reactor Vessel Internals Program is consistent with NUREG-1801 relative to monitoring and control of reactor coolant water chemistry in accordance with the EPRI Guidelines in TR-105714. RG&E is also committed to ASME Section XI, Subsection IWB (1995 Edition with 1996 Addenda). The current ASME Section XI ISI program is considered to provide reasonable assurance that aging effects will be managed such that the intended functions of reactor vessel internals components will be maintained during the license renewal period. That notwithstanding, RG&E will participate in industry activities concerning the development of augmented inspection techniques in order to visually inspect for fine cracks (0.0005 inch) and other changes in dimension in non-bolted components.

B2.1.28 Reactor Vessel Surveillance

Program Description

The Code of Federal Regulations, 10 CFR Part 50, Appendix H, requires that peak neutron fluence at the end of the design life of the vessel will not exceed 10^{17} n/cm^2 (E >1MeV), or that reactor vessel beltline materials be monitored by a surveillance program to meet the American Society for Testing and Materials (ASTM) E 185 Standard.

The existing reactor vessel material surveillance program provides sufficient material data and dosimetry to monitor irradiation embrittlement at the end of the period of extended operation, and to determine the need for operating restrictions on the inlet temperature, neutron spectrum, and neutron flux. The program was designed to meet the requirements of 10 CFR 50, Appendix H, and ASTM E-185-73. Capsules withdrawn after July 26, 1983, will be tested and the results reported in accordance with the 1982 revision of ASTM E-185 as required by 10 CFR 50, Appendix H. The program consists of six surveillance capsules (V, R, T, P, S, and N) positioned in the reactor vessel between the thermal shield and the reactor vessel wall. Capsule V was removed and tested in 1971, capsule R in 1974, capsule T in 1980, and capsule S in 1993. Capsule P is scheduled to be withdrawn at an estimated inside surface 52 effective-full-power-year fluence. Capsule N is a standby capsule and is scheduled to be withdrawn at one to two times the inside surface end-of-life fluence and stored (without testing).

Since it has been projected that the upper-shelf Charpy energy levels of the beltline weld materials may be less than 50 ft-lb at 52 EFPY of service, a low upper-shelf fracture mechanics evaluation has been performed to satisfy the requirements of Appendix G to 10 CFR Part 50. An additional capsule will be withdrawn at a neutron fluence equivalent to approximately 52 EFPY of exposure. This capsule will be stored to be made available should testing of the material specimens be required.

B2.1.28.1 Scope of Program

The program is focused on determining reactor pressure vessel loss of fracture toughness due to neutron irradiation embrittlement.

Time-Limited Aging Analysis topics related to this program include P-T Limit Curves, Upper Shelf Energy, Pressurized Thermal Shock, and LTOP setpoints. However, this program specifically addresses only Upper Shelf Energy. Other analyses are not included in the scope of this program and are addressed separately.

B2.1.28.2 Preventative Actions

This is a condition-monitoring program and therefore no preventative actions are taken.

B2.1.28.3 Parameters Monitored/Inspected

This program monitors accumulated neutron fluence from irradiated material specimens, and measures material fracture toughness and tensile strength.

B2.1.28.4 Detection of Aging Effects

Reduction in fracture toughness is the aging effect for the reactor vessel. Fracture toughness is measured against acceptance criteria to determine the extent of aging.

B2.1.28.5 Monitoring and Trending

Reactor vessel capsule neutron fluence, and Charpy V-notch upper shelf energy are monitored. This data, along with reactor vessel materials information are used as input to the analyses.

B2.1.28.6 Acceptance Criteria

Appendix G of 10 CFR Part 50 provides acceptance criteria for fracture toughness. In accordance with section IV.A.1.a of the appendix, Ginna Station has demonstrated that lower values of Charpy Upper-Shelf Energy will provide margins of safety against fracture equivalent to those required in Appendix G of Section XI of the ASME Code.

Accumulated neutron fluence for the capsule bounds the expected fluence for the beltline weld region through the period of extended operation

B2.1.28.7 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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Corrective actions are taken to prevent recurrence when the specified limits for fuel oil standards are exceeded or when water is drained during periodic surveillance. Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.28.8 Confirmation Process

Confirmation of the effectiveness of the Reactor Pressure Vessel Surveillance Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.28.9 Administrative Controls

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

B2.1.28.10 Operating Experience

Analyses that are based on the program results have been revised over time. These analyses include P-T limits, Upper Shelf Energy, Pressurized Thermal Shock, and LTOP.

The reactor vessel surveillance data for the last two capsules demonstrates that reduction in upper shelf toughness is significantly less than that predicted by regulatory guide 1.99. In addition, the last surveillance capsule tested at an estimated 3.87×10^{19} n/cm2 showed no additional reduction in upper shelf energy than the capsule tests at an estimated 1.97×10^{19} n/cm2. The reactor vessel surveillance data tend to indicate that the actual reduction in upper shelf energy has not occurred at the predicted rate. This observation provides additional assurance that the reactor vessel will continue to perform its intended function throughout the period of extended operation.

Conclusion

The Ginna Station Reactor Vessel Surveillance Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.M31, "Reactor Vessel Surveillance". The continued implementation of the Reactor Vessel Surveillance Program provides reasonable assurance that aging effects will be managed such that the intended function of the reactor pressure vessel will be maintained during the license renewal period.

B2.1.29 Selective Leaching of Materials

Program Description

This program ensures the integrity of components made of cast iron, bronze, and brass exposed to raw water (service water), treated water, or ground water at Ginna Station. The program utilizes visual inspections performed under the Periodic Surveillance/Preventive Maintenance Program (Section B2.1.23), or the One-Time Inspection Program (Section B2.1.21), to determine if selective leaching is occurring in susceptible components. The Periodic Surveillance/Preventive Maintenance Program is invoked for those potentially susceptible components which currently have a routine preventive maintenance activity. For potentially susceptible components which do not have a routine preventive maintenance activity, a one-time inspection will be performed.

Conclusion

The programs used to implement the selective leaching program at Ginna Station are similar to the program described in the GALL, with the following exception:

Hardness tests are not typically performed, although an assessment of the feasibility of performing hardness tests and the value of hardness data is made on a component specific basis.

B2.1.30 Spent Fuel Pool Neutron Absorber Monitoring

Program Description

The purpose of this program is to monitor the long-term performance of the borated stainless steel neutron absorber material used in the Ginna Station spent fuel pool.

Ginna Station also incorporates boraflex panels in the spent fuel pool. However, reliance on the neutron absorption capability of the boraflex panels was discontinued when the NRC approved License Amendment 79 on December 7, 2000. That amendment provided for reliance on soluble boron instead of the boraflex (credit for the borated stainless steel is still required).

B2.1.30.1 Scope of Program

The program monitors long term performance of the borated stainless steel (BSS) panels, using surveillance coupons comprised of the same material. The Ginna Station spent fuel pool currently has 828 locations employing (non-credited) boraflex panels, and 493 locations employing borated stainless steel.

B2.1.30.2 Preventive Actions

The Spent Fuel Pool Neutron Absorber Monitoring Program is a monitoring program only and specifies no preventive actions.

B2.1.30.3 Parameters Monitored/Inspected

The Spent Fuel Pool Neutron Absorber Monitoring Program ensures surveillance coupons are removed and evaluated as follows:

- Visual comparisons are made after the test coupons are cleaned and dried.
- Thickness measurements are taken at locations chosen to be representative of creviced/galvanically coupled areas and exposed surfaces.
- Weight measurements are taken of the test coupons using a balance capable of measuring 0.1 gram.

B2.1.30.4 Detection of Aging Effects

The BSS surveillance coupons in the spent fuel pool are periodically examined. The examinations consist of visual comparisons, thickness measurements, and weight measurements relative to reference samples that have not been exposed to the spent fuel pool environment.

B2.1.30.5 Monitoring and Trending

Parameters evaluated are recorded as directed in site specific procedures.

A schedule milestone for performing these evaluations has been determined:

- the completion of the first operating cycle following installation of the racks (milestone reached)

- the completion of every third additional operational cycle (Cycles 31, 34, 37, etc.).

B2.1.30.6 Acceptance Criteria

The comparisons are made by qualified personnel in the Ginna Station Reactor Engineering and Laboratory and Inspection Services/Chemistry sections. The values determined in the evaluation are compared to the reference values.

B2.1.30.7 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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

Any significant changes or results are to be documented in an ACTION Report per site-specific procedures and further investigated using appropriate analytical techniques. These results should also be reported to ATEA, the firm that designed, built, and installed the racks at Ginna Station

B2.1.30.8 Confirmation Process

Confirmation of the effectiveness of the Spent Fuel Pool Neutron Absorber Monitoring Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.30.9 Administrative Controls

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

B2.1.30.10 Operating Experience

A review of industry operating experience has revealed no evidence of significant age-related degradation of borated stainless steel material exposed to spent fuel pool environments.

The first examination of a coupon has been completed and the results documented in station correspondence. No evidence of degradation was found. The visual appearance of the coupons was excellent. The visual appearance and other measurements indicate that the borated stainless steel absorber panels exhibit good corrosion resistance in the spent fuel pool environment and will perform as expected over the remaining life of the racks.

Conclusion

The Spent Fuel Pool Neutron Absorber Monitoring Program is consistent with all the NUREG-1801 attributes, and is consistent with the processes required for the Boraflex Monitoring Program as described in NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M22, "Boraflex Monitoring" (Reference 3).

B2.1.31 Steam Generator Tube Integrity

Program Description

The Ginna Station Steam Generator Tube Integrity Program is a comprehensive program that incorporates the guidance of NEI 97-06 and EPRI TR-107569 and is credited for maintaining the integrity of the steam generator (SG) tubes. The program manages aging effects such as cracking due to primary water stress corrosion cracking (PWSCC), outside diameter stress corrosion cracking (ODSCC), intergranular attack (IGA), pitting, wastage, wear, fouling due to corrosion product buildup, mechanical degradation due to denting and impingement damage, and fatigue.

The program manages these aging effects/mechanisms through a balance of prevention, inspection, examination, assessment, evaluation, repair and leakage monitoring measures. The program is administered through a series of plant directives and interface procedures, as well as the plant technical specifications. Key program attributes include non-destructive examination (NDE), sludge lancing, primary and secondary water chemistry control, and primary-to-secondary leakage trending and monitoring.

B2.1.31.1 Operating Experience

Past industry and Ginna operating experience has led to sweeping changes in the programs required to maintain SG tube integrity. The NRC, EPRI and NEI have generated guidance to be used by the industry to provide effective controls. NEI has generated NEI 97-06 to incorporate lessons learned from plant operating experience and past SG tube inspection experience. Ginna Station has committed to provide a SG tube integrity program that meets these guidelines.

Ginna Station experienced many of the industry problems related to SG tube integrity in the original Westinghouse Series 44 SGs. These SGs had 28 years of operating experience. With initial startup chemistry being phosphate buffer control, these SGs experienced many of the same problems of other PWR SGs. In late 1974, the chemistry was changed over to AVT with the addition of full flow condensate demineralizers. In January, 1982, a tube rupture event occurred in the "B" SG at Ginna Station. The root cause was determined to be tube thinning due to wear caused by foreign material in the secondary side. The rate of SG tube degradation over the years, with subsequent loss of heat transfer surface due to plugging and sleeving, led to the decision to replace the SGs.

In June, 1996, the Ginna SGs were replaced with a new design, supplied by Babcock & Wilcox (B&W) International. The replacement SGs incorporate design and manufacturing improvements to reduce and/or prevent many of the problems that the industry has experienced. In addition, during the SG replacement, a modification was completed to reduce T_{AVG} in the RCS during normal operations with the intent to further prolong the life of the SG tube materials. To date, Ginna has completed SG Tube examinations and sludge lancing and secondary side foreign material/loose parts inspection in accordance with the program requirements during refuelings of 1997, 1999, and 2002 with no degradation observed. All tubes remain in service with the exception of one tube in each SG that were plugged during fabrication due to manufacturing defects. Industry experience associated with steam generators of similar design has identified tube wear due to fretting at AVBs. No evidence of wear damage has been detected in the Ginna replacement steam generators to date.

Conclusion

The Ginna Station Steam Generator Tube Integrity Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section XI.M19, "Steam Generator Tube Integrity" (Reference 3). The program provides reasonable assurance that aging effects will be managed such that the intended function of the SG tubes will be maintained during the license renewal period.

B2.1.32 Structures Monitoring Program

Program Description

The Ginna Station Structures Monitoring Program is described in an Engineering Procedure. It is a comprehensive program that was developed and implemented to meet the regulatory requirements of the maintenance rule (10 CFR 50.65, USNRC Regulatory Guide 1.160, and NUMARC 93-01). The program includes masonry walls evaluated in accordance with NRC IEB 80-11, "Masonry Wall Design" and incorporates guidance in NRC IN 87-67, "Lessons learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11" and NRC Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants." The program identifies the structures and structural components within the scope of the maintenance rule and license renewal, the performance criteria that are to be monitored, the frequency of inspections, and provides the controls to ensure that there is no loss of structure or structural component intended function.

B2.1.32.1 Operating Experience

Although the Ginna Station Structures Monitoring Program requirements have been developed and documented since 1995, plant inspection and maintenance of specific structures within the program has been on-going since initial operation. Structures such as buildings, supports, intakes, canals, etc., including roofs, block/masonry walls, liners, steel, etc. have been maintained periodically to ensure their intended function and have been upgraded consistent with regulatory requirements and industry experience.

The Ginna Station Structures Monitoring Program provides the controls necessary to ensure that any degradation can be detected and resolved prior to any loss of intended function for the period of extended operation.

Conclusion

The Ginna Station Structures Monitoring Program will be consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.S5," Masonry Wall Program," XI.S6, "Structures Monitoring Program," and XI.S7, "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants" (Reference 3). Enhancements will be made to include additional structural components consistent with the scope described above. The program will provide reasonable assurance that the aging effects from external and internal environments on structures and structural components will be managed such that their intended function will be maintained during the license renewal period.

B2.1.33 Systems Monitoring

Program Description

The Ginna Station Systems Monitoring Program is a comprehensive program which addresses aging management requirements for piping, components and equipment in systems which are within the scope of license renewal. As part of the implementation of 10 CFR 50.65 (Maintenance Rule), specific guidelines for assessing the material condition of systems, structures, and components during System Engineer walkdowns were developed. The Systems Monitoring Program is credited for managing aging effects such as loss of material, cracking, and fouling buildup for normally accessible, external surfaces of piping, tanks, and other components and equipment within the scope of License Renewal. These aging effects are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation, such as corrosion, cracking, degradation of coatings, sealants and caulking, deformation, and debris and corrosion product buildup.

B2.1.33.1 Scope of Program

The Systems Monitoring Program provides inspection requirements and guidelines for monitoring the effectiveness of maintenance in accordance with 10 CFR 50.65 (the Maintenance Rule). The scope of the program includes the accessible surfaces including insulated portions of systems, components, and equipment (including welds and bolting) which are designated as maintenance rule systems and within the scope of license renewal. The Program is based on scheduled system walkdowns, health reports, and performance monitoring and trending analysis. The Program is conducted as part of the responsibilities of System Engineers.

B2.1.33.2 Preventive Actions

The Systems Monitoring Program is primarily a condition-monitoring program. However, timely identification of aging effects before significant structural or pressure-boundary degradation occurs may be considered preventive in nature.

B2.1.33.3 Parameters Monitored/Inspected

Surface conditions of system piping and components including visible portions of insulated components, equipment, supports and closure bolting are monitored through periodic visual examinations for evidence of leakage, corrosion, cracking, coating degradation, deformation, change in material properties of flexible connections and sealants, fouling and corrosion product build-up.

B2.1.33.4 Detection of Aging Effects

Visual inspections performed during System Engineer walkdowns provide the primary means for detection and quantification of aging effects and degradation. Additional guidance for identifying and evaluating the evidence of degradation will be included in appropriate plant procedures. Degradation that is deemed "Unacceptable" will be addressed using the Ginna Station Corrective Action process. The Systems Monitoring Program is designed for early detection of age-related degradation prior to system or component failure.

Accessible portions of maintenance rule and license renewal systems are required to be walked down once per quarter. Walkdowns are scheduled and performed so that the entire system is fully inspected within one operating cycle.

B2.1.33.5 Monitoring and Trending

Detailed system/equipment material condition inspections will be performed according to instructions in pertinent administrative procedures. Data from inspections performed during walkdowns is documented, trended and evaluated. The frequency of material condition inspections may be adjusted as necessary based on inspection results and industry experience.

B2.1.33.6 Acceptance Criteria

Administrative procedures will be enhanced to include visual inspection acceptance criteria. Guidance for the assessment of surface corrosion that includes consideration of design margins will be provided in the enhanced procedures. Additional guidance on evaluation of protective coatings will be included. Detection of pressure-boundary leakage requires assessment and appropriate correction.

B2.1.33.7 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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.33.8 Confirmation Process

Confirmation of the effectiveness of the System Monitoring Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.33.9 Administrative Controls

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

B2.1.33.10 Operating Experience

System inspection requirements that have been in effect at Ginna Station since the mid 1990's in support of the Maintenance Rule have proved to be effective in maintaining the material condition of plant systems. A significant number of corrective actions have been processed as a result of System Engineer walkdowns. The Systems Monitoring Program will also be continually re-assessed and upgraded based on industry and plant-specific operating experience reviews.

Conclusion

The Ginna Station Systems Monitoring Program, with the enhancements identified above, has been evaluated using the generic program attributes identified in Appendix A of the SRP. The continued implementation of the Program provides reasonable assurance that aging effects will be managed such that the intended function of systems and components within the scope of license renewal will be maintained during the extended period of operation. In addition to specific enhancements identified in the attributes described above, additional systems/components, consistent with the scope of license renewal, will be included in a future revision of appropriate Ginna Station Procedures.

B2.1.34 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)

Program Description

The potential exists for a loss of fracture toughness due to thermal aging of cast austenitic stainless steel (CASS) components. An evaluation of the susceptibility of CASS components at Ginna Station was made, based on the casting method, molybdenum content, and percent ferrite. It was determined that the CASS RCS elbows were susceptible to loss of fracture toughness due to thermal aging. A plant-specific flaw tolerance evaluation was conducted, and documented in 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. The evaluation concluded that adequate fracture toughness exists for the RCS loop, including the cast elbows, for the period of extended operation (60 years).

A separate evaluation was made for the reactor coolant pump casings. In WCAP-15873, "A Demonstration of the Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of R.E. Ginna Nuclear Power Plant for the License Renewal Program", May 2002, it was concluded that the primary loop pump casings are qualified to item (d) of ASME Code Case N-481 for the period of extended operation (60 years).

B2.1.35 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)

Program Description

The reactor vessel internals receive a visual inspection in accordance with ASME Code Section XI, Subsection IWB, Category B-N-3. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling of CASS reactor vessel internals.

Ginna Station has performed an evaluation to determine potentially susceptible components of the internals. No reactor vessel internal components made of CASS that serve a license renewal intended function have been identified.

Therefore, this program is not applicable to Ginna Station.

B2.1.36 Thimble Tubes Inspection

Program Description

This program manages the integrity of the incore neutron monitoring thimble tubes, which serve as a portion of the reactor coolant pressure boundary. As discussed in NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors," July 26, 1988, thimble tube wall-thinning can occur as a result of flow-induced vibration. This wear damage is detected at locations associated with geometric discontinuities or area changes along the reactor coolant flow path, such as areas near the lower core plate, the core support forging, the lower tie plate, and the vessel penetrations.

Periodic assessment of thimble tube wear, and corrective actions as needed, form the basis for this program.

B2.1.36.1 Scope of Program

All thirty-six thimble tubes are within the scope of this inspection program.

B2.1.36.2 Preventive Actions

As noted in operating experience below, the replacement of tube G-6 with chrome plating at the wear area constitutes a preventive action. In addition flushing of the tubes during refueling outages is also considered preventive in nature.

Eddy current examinations are performed on a periodicity consistent with the severity of wear damage for each thimble tube. When wall loss in a tube exceeds 55%, but less than 65%, the tube is repositioned such that wear is redistributed, or other corrective action is taken.

The eddy current examinations themselves are inspection/verification activities, and are thus not considered preventive.

B2.1.36.3 Parameters Monitored/Inspected

The eddy current examinations determine the wall thickness of the thimble tubes, allowing an assessment of the wear, and wear rate, of each tube in each location.

B2.1.36.4 Detection of Aging Effects

Thimble tube inspections are conducted using a methodology specified in a Ginna Station plant-specific procedure. This procedure requires the use of a Zetec MIZ-18 Multifrequency Eddy Current Testing System. These inspections provide indication of tube wear, and tube wear rate.

B2.1.36.5 Monitoring and Trending

Based on the results of a plant-specific analysis, examination results are compared to an upper allowable limit of 65% through-wall wear.

Eddy current examinations performed in 1988, 1989, 1990, 1991, and 1992 provided a basis for establishing the wear rates, and thus the inspection intervals, for thimble tubes. Based on those results, the inspection frequency and acceptance criteria are:

- previous indication 10% to less than 45% - every third refueling outage (approximately once every 4.2 years)
- previous indication 45% to less than 55% - every other refueling outage (approximately once every three years)
- previous indication 55% or greater - perform corrective action, if support plate wear is the suspected cause. For other indications, corrective action

will be taken at 65% or greater. Future inspection frequency will be every other or every third outage, as stated above.

- previous inspection never exceeded 10% through-wall - no specified frequency. Future inspections will be based on a Ginna Station periodic assessment.

B2.1.36.6 Acceptance Criteria

The acceptance criteria is provided in (Section B2.1.36.5).

B2.1.36.7 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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B2.1.36.8 Confirmation Process

Confirmation of the effectiveness of the Thimble Tube Inspection Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B2.1.36.9 Administrative Controls

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

B2.1.36.10 Operating Experience

Thimble tube wear in Westinghouse reactors was documented in NRC IN 87-44, "Thimble Tube Thinning in Westinghouse Reactors," and NRC Bulletin 88-09. In response to these notifications, eddy current examination of thimble tubes was performed annually from 1988 to 1992 at Ginna Station.

In 1990, thimble tube G-6 had indication of wear greater than 55%. Corrective action was taken by repositioning (moving worn areas away from the lower support plate by 1 - 2") the tube.

Three other thimble tubes had indications noted in the 1997 examination that resulted in the need for corrective action (ACTION Report 97-1889). All four thimble tubes were replaced during the 1999 refueling outage. One thimble had an indication of intergranular attack. The conduit water was sampled, and analysis showed the presence of chlorides, fluorides, and sulfates in concentrations significantly above RCS water. These conduits were flushed during the thimble tube replacement. All other thimble tube conduits were flushed during the 2000 refueling outage.

During the 2000 refueling outage, inspection of tube G-6 again indicated degradation due to flow-induced vibration. This tube was replaced with a chrome-plated tube during the 2002 refueling outage.

Conclusion

Although the Ginna Station Thimble Tube Inspection Program is a plant-specific program consistent with our commitments to Bulletin 88-09, the attributes associated with the ASME Section XI, Subsection IWB, IWC, IWD Inservice Inspection Program (Section B2.1.2) are met.

B2.1.37 Water Chemistry Control

Program Description

The Water Chemistry Control Program mitigates damage caused by aging effects by controlling the internal environment of components in the primary, borated, and secondary water systems. Aging effects managed by this program include loss of material due to general, pitting and crevice corrosion, microbiologically influenced corrosion (MIC), stress corrosion cracking (SCC), and fouling due to corrosion product buildup. The program relies on monitoring and control of water chemistry based on the EPRI guidelines in TR-105714 (Reference 7) for primary systems chemistry and TR-102134 (Reference 8) for secondary systems chemistry.

For low-flow or stagnant portions of a system, a one-time inspection of selected components at susceptible locations provides verification of the effectiveness of the Water Chemistry Control Program. This inspection is covered within the scope of the One-time Inspection Program (Section B2.1.21). No verification inspections are required for intermediate and high flow regions.

The aging effects are managed by controlling concentrations of known detrimental chemical species such as chlorides, fluorides, sulfates and dissolved oxygen below the levels known to cause degradation. The program includes specifications for chemical species, sampling and analysis frequencies, and corrective actions for control of water chemistry.

B2.1.37.1 Operating Experience

The EPRI guideline documents have been developed based on industry experience and have been shown to be effective over time with their widespread use. Industry operating experience related to specific issues have been incorporated into subsequent revisions of these guidelines upon which the plant specific program is based.

A review of plant specific operating experience indicates that Ginna Station has experienced a single Level 3 excursion, which was in the Reactor Coolant System oxygen concentration after the Cycle 17 refueling startup. No other Level 3 excursions were found. An independent assessment of the primary and secondary chemistry programs at Ginna Station is routinely performed to confirm that the program is maintained within plant specifications and industry guidelines. Recommendations from these assessments have been used to improve plant chemistry and overall plant operations.

In 1996, Ginna station replaced the original Westinghouse model 44 steam generators with BWI replacement steam generators incorporating design enhancements and Inconel 690 TT tubing. The new design is less susceptible to many of the aging effects managed by this program. In addition, installation of a new reactor vessel closure head incorporating design enhancements and Inconel 690 TT penetrations is planned for the 2003 refueling outage.

Conclusion

The Ginna Station Water Chemistry Control program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Sections XI.M2, "Water Chemistry Control" (Reference 3), however, plant procedures and guidelines for water chemistry control are based on Revision 5 of EPRI TR-102134 (Reference 8) and Revision 4 of TR-105714 (Reference 7).

The existing water chemistry control program has demonstrated that the aging effects associated with applicable components have been adequately managed in the current operating term. Plant specific experience has been used to enhance the program over time, consistent with the requirements of the corrective action program. Further enhancements as described in this program will provide additional assurance that these components will perform their intended functions in accordance with the current licensing basis during the period of extended operation.

B3.0 TIME-LIMITED AGING ANALYSES SUPPORT ACTIVITIES

B3.1 Environmental Qualification Program

Program Description

The Nuclear Regulatory Commission (NRC) has established Environmental Qualification of Electrical Equipment requirements in 10 CFR Part 50, Appendix A, Criterion 4, and 10 CFR 50.49. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments (that is, those areas of the plant that could be subject to the harsh environmental effects of a loss of coolant accident [LOCA], high energy line breaks [HELBs] or post-LOCA radiation) are qualified to perform their safety function in those harsh environments after the effects of inservice aging. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

The Environmental Qualification (EQ) program manages component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods, for those components within the scope of the rule. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the age limits established in the evaluation.

Reanalysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(e) may be performed as part of an EQ program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed in NUREG-1800 and section 4.4 of the License Renewal Application.

B3.1.1 Scope of Program

The Ginna Station EQ program applies to certain electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in 10 CFR 50.49.

B3.1.2 Preventive Actions

10 CFR 50.49 does not require actions that prevent aging effects. Ginna Station EQ program actions that could be viewed as preventive actions include (a) establishing service conditions for components to extend equipment qualified life and (b) requiring specific installation, inspection, monitoring or periodic maintenance actions to maintain component aging effects within the bounds of the qualification basis.

B3.1.3 Parameters Monitored/Inspected

EQ component qualified life is based on testing and analysis not on condition or performance monitoring. However, pursuant to Regulatory Guide 1.89, Rev. 1, such monitoring programs are an acceptable basis to modify a qualified life through reanalysis. Monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.

B3.1.4 Detection of Aging Effects

10 CFR 50.49 does not require the detection of aging effects for in-service components. Monitoring of environmental conditions such as temperature in the immediate vicinity of EQ equipment is used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.

B3.1.5 Monitoring and Trending

10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters to manage the effects of aging. However, Ginna Station EQ program actions do include monitoring how long qualified components have been installed, how often they are operated/energized, and what discrete environmental conditions such as temperature and radiation exist. Such monitoring is used to ensure

that select components are within the bounds of their qualification bases, or to modify the qualified life.

B3.1.6 Acceptance Criteria

10 CFR 50.49 acceptance criteria are that an inservice EQ component is maintained within the bounds of its qualification basis, including (a) its established qualified life and (b) continued qualification for the projected accident conditions. 10 CFR 50.49 requires refurbishment, replacement, or requalification prior to exceeding the qualified life of each installed device. Where monitoring has been used to modify a component qualified life, Ginna-specific acceptance criteria for operating in those conditions were established.

B3.1.7 Corrective Actions

If an EQ component is found to be outside the bounds of its qualification basis, corrective actions are implemented in accordance with the requirements of 10 CFR 50, App. B "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants", as described in the Ginna "Quality Assurance Program for Station Operation" (ND-QAP). When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions. When an emerging industry aging issue is identified that affects the qualification of an EQ component, the affected component is evaluated and appropriate corrective actions are taken, which may also include changes to the qualification bases and conclusions.

B3.1.8 Confirmation Process

Confirmation of the effectiveness of the Environmental Qualification Program is accomplished in accordance with the Ginna Station Corrective Action Program, the site QA procedures, review and approval processes and administrative controls, which are implemented in accordance with the requirements of 10 CFR 50, App. B.

B3.1.9 Administrative Controls

The EQ programs is implemented through the use of station policy, directives, and procedures. The EQ program will continue to comply with 10 CFR 50.49 throughout the renewal period, including development and maintenance of qualification documentation demonstrating reasonable assurance that a component can perform required functions during harsh accident conditions. EQ program documents identify the applicable environmental conditions for the component locations. EQ program qualification files are maintained at the plant site in an auditable form for the duration of the installed life of the component. EQ program documentation is controlled under the station's quality assurance program.

B3.1.10 Operating Experience

As a result of 10 CFR 50.49, Ginna Station installed extensive new environmentally qualified electrical equipment, in accordance with the "DOR Guidelines". The type of equipment replaced or installed included transmitters, level switches, electrical cable, solenoid valves, connectors, splices, and LVDTs. Upgrades were also made to electrical motors, valve activators, and electrical penetrations.

Ginna maintains cognizance of emerging issues in EQ and aging management by actively participating in industry forums, including the Nuclear Utility Group on Equipment Qualification.

Conclusion

The Ginna Station Environmental Qualification Program provides reasonable assurance that compliance with 10 CFR 50.49 is maintained. EQ components are designed and maintained to perform their intended functions in a postulated post-accident, environment, following the effects of inservice aging for the period of extended operation.

B3.2 Fatigue Monitoring

Program Description

The Fatigue Monitoring Program is a newly incorporated program that is consistent with the NRC Generic Aging Lessons Learned (GALL) Report, Section X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary". The Fatigue Monitoring Program is a confirmatory program that monitors loading cycles due to thermal and pressure transients for selected critical components. The program provides an analytical basis for confirming that the number of cycles established by the analysis of record will not be exceeded before the end of the period of extended operation. The effects of reactor coolant environment are considered through the evaluation of the seven component locations identified in NUREG/CR-6260 using the appropriate environmental fatigue factors. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon or low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels, or other environmental fatigue factors appropriate to the material.

B3.2.1 Scope of Program

The scope of the Fatigue Monitoring Program includes those plant systems and components for which a cyclic or fatigue design basis exists. The specific systems and components included within the scope of the Fatigue Monitoring Program are identified below:

Reactor Pressure Vessel Closure Studs

Reactor Pressure Vessel Primary (Inlet and Outlet) Nozzles

Reactor Pressure Vessel at Core Support Pad

Steam Generator Tubesheet

Cold Leg (Accumulator) Safety Injection Nozzle

Pressurizer Upper Shell

Pressurizer Spray Nozzle

Pressurizer Surge Line Nozzle

Hot Leg Surge Line Nozzle

Pressurizer Surge Line

Pressurizer Heater Well Penetration

Reactor Coolant Piping Charging System Nozzles

Residual Heat Removal Hot Leg Suction Nozzles

Residual Heat Removal System Class 1 Piping

B3.2.2 Preventive Actions

The Fatigue Monitoring Program includes preventive measures to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary, and will provide adequate margin against fatigue cracking of these components due to anticipated cyclic strains.

Tracking of operating transient cycles and maintaining the fatigue usage factor below the design code limit of 1.0, including the effects of reactor water environment, will provide adequate margin against fatigue cracking of reactor coolant system components due to anticipated cyclic strains.

B3.2.3 Parameters Monitored, Inspected, and/or Tested

The Fatigue Monitoring Program monitors plant transients that cause cyclic strains and are significant contributors to fatigue damage or crack growth. The Fatigue Monitoring Program consists of automated cycle counting to count the number of plant transients that cause significant fatigue damage. Fatigue usage factors are tracked for bounding component locations of the reactor coolant pressure boundary.

B3.2.4 Detection of Aging Effects

The Fatigue Monitoring Program provides for periodic updates of the plant cycle count and fatigue usage calculations. The metal fatigue aging effect will be monitored using FatiguePro™, which is an EPRI software product for plant transient monitoring and fatigue usage and fatigue crack growth calculations. Plant operating cycles will be tracked against design limits. Fatigue usage factors will be computed on an on-going basis for bounding components using plant instrument data.

B3.2.5 Monitoring and Trending

The Fatigue Monitoring Program includes monitoring the number and severity of plant design transients and an on-going fatigue analysis of a sampling of component locations whose level of metal fatigue is expected to be most adversely affected by the combined effects of plant cycles and reactor water environment. The monitored population includes each of the component locations identified in NUREG/CR-6260 for older vintage Westinghouse plants, as well as others listed in Section B3.2.1 above.

The Fatigue Monitoring Program will ensure that the extent of the fatigue aging effect is quantifiable on an on-going basis and that alternative or mitigative actions can be taken before there is a loss of any component's intended function. The program monitors operating transients to-date, calculates fatigue usage factors to-date, and allows corrective measures to be implemented ahead of time to ensure that structural margins required by the Codes used in the original plant design are maintained throughout the operating life of the plant. The annual recording and assessment frequency ensures that normal operating transients that might occur during the plant operational period will not compromise these limits. The evaluation locations have been chosen to ensure that locations that might approach acceptance limits will be monitored. The program also includes provisions to identify deviations from expected usage factor accumulation so that appropriate corrective actions can be taken before structural margins are degraded to unacceptable levels.

B3.2.6 Acceptance Criteria

The acceptance criterion consists of maintaining the fatigue usage less than or equal to the design code allowable limit of 1.0, considering environmental fatigue effects. The acceptance criteria will ensure that original structural margins considered in the plant design are maintained throughout the operating period.

B3.2.7 Corrective Actions

Corrective actions 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 in the Ginna Station UFSAR. Provisions for timely evaluation of adverse conditions and implementation of any corrective actions required, including root cause analyses 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 site-specific procedures.

The Fatigue Monitoring Program provides for corrective actions to prevent the fatigue usage factor from exceeding the design code limit of 1.0 during the period of extended operation. The Fatigue Monitoring Program utilizes FatiguePro™ both to perform an analysis of each monitored component location using actual plant data and to provide the basis for proactive action to maintain the fatigue usage factors below Code limits. Corrective actions include a review of additional affected component locations.

For component locations for which it can not be demonstrated that the fatigue usage factor remains below the design code limit of 1.0 during the period of extended operation, corrective actions can include a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded, repair or replacement of the component, or managing the effects of fatigue by an inspection program that has been reviewed and approved by the NRC (e.g. periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).

B3.2.8 Confirmation Process

Confirmation of the effectiveness of the Fatigue Monitoring Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B3.2.9 Administrative Controls

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

B3.2.10 Operating Experience

The Fatigue Monitoring Program includes reviews of both industry and plant-specific operating experience regarding fatigue cracking for applicability to Ginna. These on-going reviews will be considered when selecting additional monitored components. As needed, additional inspections, or other analytical programs, will be used to ensure that unacceptable fatigue cracking does not occur, due to both anticipated and unanticipated transients.

Several industry issues have arisen which have provided knowledge about the propensity for fatigue cracking in Class 1 and 2 components. Starting in 1979, several PWRs experienced main feedwater piping and nozzle fatigue cracking resulting from thermal stratification cycling during conditions of low flow and hot standby. NRC Bulletin 79-13 dealt with fatigue cracking of steam generator feedwater nozzles (Class 2 component). As reported in 1980, through-wall cracking in the area of the auxiliary feedwater connection to the main feedwater piping occurred at Ginna Station. The J-tube design was implemented in the early 1980's. The currently installed replacement steam generators for Ginna include main feedwater nozzle features that minimize thermal stratification effects. Thus, the main feedwater nozzles are not included in the monitored locations.

NRC Bulletin 88-08 addressed the potential for fatigue cracking in normally stagnant piping systems attached to the reactor coolant system. A number of cold leg safety injection pipe cracking incidents (Farley-Unit 2 in 1987 and Sequoyah Unit 2 in 1996) and thermal sleeve cracking incidents (Trojan and McGuire Unit 1) have occurred at Westinghouse plants. This piping and nozzles were associated with the 10-inch accumulator line connection to the cold legs. The Fatigue Monitoring Program includes the nearby cold leg safety injection nozzles as monitored component locations. These locations are more fatigue-sensitive than the location evaluated in NUREG/CR-6260 and are considered to bound the NUREG/CR-6260 location. In addition, NDE of the most sensitive locations at Ginna Station has demonstrated that damage would not be expected due to inleakage-induced thermal transients.

Conclusion

The Ginna Station Fatigue Monitoring Program is consistent with GALL Section X.M1 "Metal Fatigue of Reactor Coolant Pressure Boundary" aging management program. However, corrective actions at Ginna Station may include management of the effects of fatigue by an inspection program (e.g. periodic non-destructive examination of the affected locations at defined inspection intervals) should an appropriate inspection technique and interval be developed and subsequently approved by the NRC.

B3.3 Concrete Containment Tendon Pre-stress

Program Description

In order to ensure the adequacy of prestressing forces in prestressed concrete containments during the extended period of operation, a Time-Limited Aging Analysis was performed. The results of this analysis indicated that continued monitoring and potential retensioning of the containment tendons may be necessary to ensure that the prestressing forces remain above the minimum required value for all tendons.

The aging management program (AMP) consists of an assessment of the results of inspections performed in accordance with the requirements of Subsection IWL of the ASME Section XI Code (Reference 19), as supplemented by the requirements of 10 CFR 50.55a(b)(2)(ix) or (viii) in the later amendment of the regulation. The assessment related to the adequacy of the prestressing force will consist of the establishment of (1) acceptance criteria and (2) trend lines. The acceptance criteria are developed consistent with the methodology of NRC Regulatory Guide 1.35.1 (Reference 21), and will normally consist of predicted lower limit (PLL) and the minimum required prestressing force, also called minimum required value (MRV). NRC Information Notice IN 99-10 (Reference 20) provides guidance for constructing the trend line. The goal is to keep the trend line above the PLL because, as a result of any inspection performed in accordance with ASME Section XI, Subsection IWL, if the trend line crosses the PLL, the existing prestress in the containment could go below the MRV soon after the inspection and would not meet the requirements of 10 CFR 50.55a(b)(2)(ix)(B) or 10 CFR 50.55a(b)(2)(viii)(B).

B3.3.1 Scope of Program:

The program addresses the assessment of containment prestressing force.

B3.3.2 Preventive Actions:

Maintaining the prestress above the MRV, as described under program description above, will ensure that the structural and functional adequacy of the containment are maintained.

B3.3.3 Parameters Monitored/Inspected:

The parameters to be monitored are the containment prestressing forces in accordance with requirements specified in Subsection IWL of Section XI of the ASME Code, as incorporated by reference in 10 CFR 50.55a.

B3.3.4 Detection of Aging Effects:

This program detects the loss of containment prestressing forces.

B3.3.5 **Monitoring and Trending:**

The estimated and measured prestressing forces are plotted against time and the PLL, MRV, and trending lines developed for the period of extended operation.

B3.3.6 **Acceptance Criteria:**

The prestressing force trend lines indicate that existing prestressing forces in the containment would not be below the MRVs prior to the next scheduled inspection, as required by 10 CFR 50.55a(b)(2)(ix)(B) or 10 CFR 50.55a(b)(2)(viii)(B).

B3.3.7 **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" as described in the "Quality Assurance Program for Station Operation" (ND-QAP). Provisions for timely evaluation of adverse conditions and implementation of required corrective actions, including root cause determinations and prevention of recurrence, are included in the Ginna Station Corrective Action Program.

B3.3.8 **Conformation Process:**

Confirmation of the effectiveness of the Concrete Containment Tendon Prestress Program is accomplished in accordance with the Ginna Station Corrective Action Program, site Quality Assurance (QA) procedures, review and approval processes and administrative controls which are implemented in accordance with the requirements of 10 CFR 50, Appendix B.

B3.3.9 **Administrative Controls:**

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

B3.3.10 **Operating Experience:**

Ginna station retensioned 23 of the 160 vertical tendons 1000 hours after initial prestressing. Subsequent tests identified that tendon lift-off forces were generally lower than the predicted values. An investigation was started to determine the reason for the accelerated loss of lift-off forces. Prior to completing the investigation, Ginna retensioned the 137 tendons that were not originally retensioned. The investigation concluded that stress relaxation of the tendon wires was the only significant cause for the lower-than-predicted tendon forces. To quantify these findings, RG&E initiated a tendon stress relaxation test program that was conducted at the Fritz Engineering Laboratory of Lehigh University.

The Time-Limited Aging Analysis for the Evaluation of Loss of Prestress in Containment Tendons concluded that the initial retensioned set of 23 tendons should be retensioned prior to the end of the current licensing period to ensure that prestressing forces remain above the minimum required value in the period of extended operation.

Conclusion:

The Ginna Station Concrete Containment Tendon Prestress Program is consistent with NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Section X.S1. The program provides reasonable assurance that the aging effect of loss of containment prestressing forces will be managed such that the tendon intended functions will be maintained during the period of extended operation.

Appendix B References

1. NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, July 2001.

2. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants"

3. NUREG-1801, *Generic Aging Lessons Learned (GALL) Report,* U.S. Nuclear Regulatory Commission, July 2001.

4. NUREG/CR-6260 (INEL-95/0045), "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," U.S. Nuclear Regulatory Commission, March 1995.

5. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"

6. ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants".

7. EPRI TR-105714, "PWR Primary Water Chemistry Guidelines," Revision 4, Electric Power Research Institute.

8. EPRI TR-102134, "PWR Secondary Water Chemistry Guidelines," Revision 5, Electric Power Research Institute.

9. ASME Boiler and Pressure Vessel Code, Section XI, Appendix L "Operating Plant Fatigue Assessment," 1995 Edition.

10. United States Nuclear Regulatory Commission, Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," 6/22/1988, including Supplements 1, 2, and 3, dated 6/24/88, 8/4/88 and 4/11/89

11. NRC Information Notice 97-019, "Safety Injection System Weld Flaw at Sequoyah Nuclear Power Plant, Unit 2," 4/18/97

12. NRC Information Notice 82-030, "Loss of Thermal Sleeves in Reactor Coolant System Piping at Certain Westinghouse PWR Power Plants," 7/26/82

13. EPRI TR-107396, Closed Cooling Water Chemistry Guidelines, Electric Power Research Institute, Palo Alto, CA, November 1997.

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

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

16. SAND 96-0344, Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations, Sandia National Laboratories for the U. S. Department of Energy, September 1996.

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

18. NEI 95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 -The License Renewal Rule, Rev. 3, Nuclear Energy Institute, March 2001.

19. ASME Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Plants, 1992 Edition with 1992 Addenda; 1995 Edition with 1996 Addenda, The ASME Boiler and Pressure Vessel Code, The American Society of Mechanical Engineers, New York, NY.

20. NRC Information Notice 99-10, Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments, U. S. Nuclear Regulatory Commission, April 1999.

21. NRC Regulatory Guide 1.35.1, Determining Prestressing Forces for Inspection of Prestressed Concrete Containments, U. S. Nuclear Regulatory Commission, July 1990.

APPENDIX C

(Not Used for This Application)

APPENDIX C
Contents

C1.0 APPENDIX C - NOT USED

Appendix C is not used in this application.

APPENDIX D

TECHNICAL SPECIFICATION CHANGES

APPENDIX D
Contents

D2.0 APPENDIX D - TECHNICAL SPECIFICATIONS CHANGES

10 CFR 54.22, requires that an application for license renewal include any Technical Specification changes, or additions that are necessary to manage the effects of aging during the period of extended operation. A review of the information provided in this License Renewal Application and the Ginna Station Technical Specifications confirms that no changes to the Technical Specifications are necessary.