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Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
License Renewal Safety Evaluation Report Open Item Responses
(TAC No. MA8054)

Gentlemen:

By letter dated January 10, 2001 (1CNA010106), the NRC issued the Arkansas Nuclear One, Unit 1 (ANO-1) License Renewal Safety Evaluation Report (SER) with open items. Attachment A provides responses to the six open items. Also, by letter dated January 31, 2001 (1CAN010103), Entergy Operations committed to provide an update to the ANO-1 Safety Analysis Report (SAR) Supplement (Appendix A of the ANO-1 License Renewal Application (LRA)) with the SER open item responses. Attachment B describes the changes to ANO-1 LRA Appendix A (SAR Supplement) which is also attached. This submittal provides a commitment to include additional fire protection equipment within the scope of license renewal as discussed in response to Open Item 2.3.3.2.2-1. Should you have further questions, please contact me.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on March 14, 2001.

Very truly yours,

Jimmy D. Vandergrift
Director, Nuclear Safety Assurance

JDV/nbm
Attachments

A082

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Attachment A
Open Item Responses

Open Item 2.3.2.6.2-1 - The ANO-1 SAR, Section 6.2.2.1, identifies an in-line flow orifice as being necessary to ensure proper sodium hydroxide injection rate for pH control. This flow orifice is not identified as a component of the sodium hydroxide system that is subject to an aging management review for its intended function of flow control, refer to Table 3.3-6, of the LRA.

In addition to the pressure boundary function, the function of *flow control* has been added to the site aging management review report for the flow orifices in the sodium hydroxide system. The orifices are constructed of stainless steel and have cracking and loss of material as aging effects that require management. The inspections of the sodium hydroxide system stainless steel components will manage the aging effects for these orifices. The inspections will be completed as a part of the new Augmented Inspections for the American Society of Mechanical Engineers (ASME) Section XI Inservice Inspections identified in the response to request for additional information (RAI) 3.3.3.3-14 as documented in correspondence dated September 12, 2000 (1CAN090004).

Open Item 2.3.3.2.2-1 - The applicant does not include fire protection jockey pump, carbon dioxide systems, fire hydrants, the water supply to the low level radwaste building fire protection system, and the piping to the manual hose station as being within the scope of license renewal, and subject to an aging management review. The Staff requested additional justification for the exclusion of these components but, on the basis of the additional information provided to the NRC to date, the applicant has not provided sufficient justification for the exclusion of these components. Sufficient justification needs to be provided, or the fire protection jockey pump, carbon dioxide systems, fire hydrants, the water supply to the low level radwaste building fire protection system, and the piping to the manual hose station (located downstream of FS-43) need to be included within the scope of license renewal, and subject to an aging management review.

By letter dated November 2, 2000 (1CAN110001), Entergy Operations provided clarifications pertaining to requests for additional information related to fire protection. A detailed justification for each of these individual components being excluded from the scope of license renewal was provided. A meeting was held between Entergy Operations and the NRC Staff on March 8, 2001, to discuss these components. Entergy Operations provided information sufficient to demonstrate that the carbon dioxide systems, the water supply to the low level radwaste building fire protection system, and the piping to the manual hose station are not necessary for compliance with 10CFR50.48. However, Entergy Operations agrees to include the fire protection jockey pump and the fire hydrants as components necessary to comply with 10CFR50.48, and therefore, these components will be included within the scope of license renewal [commitment]. These components have material and environment combinations similar to the other components considered

in the fire protection aging management review and will be added to the existing aging management programs.

Open Item 3.3-1 - The Staff reviewed the applicant's summary descriptions of the aging management programs, and the evaluations of the time-limited aging analyses (TLAAs) provided by the applicant in Appendix A, "Safety Analysis Report Supplement," of the LRA, to ensure that they are consistent with the requirements of 10CFR54.21(d). The Staff identified a number of summary description of aging management programs and TLAA evaluation that need addition information to meet the intent of 10CFR54.21(d). The additional information needed include the following:

- **SAR Item 3.3.1.2.3 - A summary description of the quality assurance aging management program is needed in Appendix A of the LRA. This summary description that adequately describes the corrective action program (specifically describes corrective actions, the confirmation process, and the administrative controls consistent with 10CFR Part 50, Appendix B) as it applies to license renewal needs to be included in the SAR supplement in accordance with 10CFR54.21(d). In a letter to the NRC dated October 3, 2000, the applicant states that this description will be submitted with its annual update.**

  See Section 16.0 of the revised ANO-1 SAR Chapter 16 (attached).

- **SAR Item 3.3.1.3.3 - Based on the information provided in the letter to the NRC dated September 7, 2000, the applicant should indicate in Appendix A, "Safety Analysis Report Supplement," of the LRA, that the maintenance rule program applies only to external surfaces of the structures and components that use the aging management program.**

  See Section 16.2.13 of the revised ANO-1 SAR Chapter 16 (attached).

- **SAR Item 3.3.1.4.1.3 - On the basis of the information provided in the letter to the NRC dated September 7, 2000, in Appendix A, "Safety Analysis Report Supplement," of the LRA, the applicant needs to state that the buried pipe inspection program also applies to the buried piping in the fire protection systems that are within scope of license renewal and subject to an aging management review.**

  As identified in the response to RAI 3.3.4.3.1-2(d) contained in the September 12, 2000, correspondence (1CAN090004), the fire protection system contains buried piping that is within the scope of license renewal. As identified in the RAI 3.3.4.3.1-2(d) response, the aging management program that will manage this aging effect is the Fire Suppression Water Supply System Surveillance. The

listing of systems covered by the Buried Pipe Inspection Program is correct, and no change is required to Section 16.1.1 of the SAR supplement.

- <u>SAR Item 3.3.1.4.2.3</u> - **On the basis of the information provided in Table 3.4-10 of the LRA, the applicant needs to state, in Appendix A, "Safety Analysis Report Supplement," of the LRA, that fouling is also an aging effect managed by the heat exchanger monitoring program. In addition, the applicant needs clarify that fouling on the service water side of the decay heat removal heat exchangers is managed through two aging management programs, the heat exchanger monitoring program and the service water integrity program.**

  As identified in the response to RAI 3.3.4.3.2.10-2 in correspondence dated September 12, 2000 (1CAN090004), the Heat Exchanger Monitoring Program does *not* address fouling. The Heat Exchanger Monitoring Program will inspect heat exchangers to the extent required to ensure seismic qualification is maintained, but it is not intended to be the program to monitor for fouling. Fouling will be detected by other programs such as the Service Water Integrity Program or system surveillance testing. The SAR Section 16.1.3 description of the Heat Exchanger Monitoring Program correctly identifies the aging effects of cracking and loss of material that could result in degradation of the seismic qualification of the heat exchangers and does not require a revision.

- <u>SAR Item 3.3.1.4.3.3</u> - **On the basis of the information provided in the letter to the NRC dated September 12, 2000, in Appendix A, "Safety Analysis Report Supplement," the applicant needs to indicate that the only components in the reactor building isolation system to which the wall thinning inspection program is applied are the carbon steel components that are associated with reactor building penetration numbers 51 and 59.**

  The response to RAI 3.3.4.3.2.9-2 in correspondence dated September 12, 2000 (1CAN090004), was a specific response regarding which carbon steel components of the chilled water system credited the Wall Thinning Inspection Program. The response was limited to the chilled water components of penetrations 51 and 59. The response to this RAI did not include other reactor building isolation system carbon steel components that credit the Wall Thinning Inspection Program. These other penetrations are already listed in the program description in Appendix B of the LRA (Section 3.7). The listing of penetrations covered by the Wall Thinning Inspection Program is correct, and no change is required to Section 16.1.7 of the SAR supplement.

- <u>SAR Item 3.3.2.4.3</u> - **For the control rod drive mechanism nozzle and other vessel closure penetrations inspection program, if an applicant-specific inspection program is determined to be necessary the applicant will analyze**

and evaluate axial flaws using NUMARC acceptance criteria, and will address circumferential flaws with the NRC on a case-by-case basis.

See Section 16.2.7 of the revised ANO-1 SAR Chapter 16 (attached).

- SAR Item 3.3.3.3 - **The applicant needs to add a one-time inspection to detect cracking or wall thinning of piping and fittings in the sodium hydroxide system to the summary description of the Augmented Inspection program in Section 16.2.3.7 of Appendix A of the LRA. Upon including this summary description in the final SAR Supplement for license renewal, the applicant will have met the requirements of 10CFR54.21(d) for this aging management program.**

  See Section 16.2.3.7 of the revised ANO-1 SAR Chapter 16 (attached).

- SAR Item 3.3.7.4 - **Upon identifying an acceptable aging management programs for managing a reduced insulation to ground, and potential electrical failure of buried (inaccessible) medium-voltage [cables] due to moisture intrusion, water treeing, and contamination, the applicant needs to include an adequate summary description of the aging management programs in the SAR Supplement consistent with 10CFR54.21(d).**

  See Section 16.1.2 of the revised ANO-1 SAR Chapter 16 (attached).

- SAR Item 4.3.4 - **A summary description that adequately describes the applicant's proposed program to address environmental effects on fatigue is needed to meet the requirements of 10CFR54.21(d).**

  See Section 16.3.2 of the revised ANO-1 SAR Chapter 16 (attached).

- SAR Item 4.5.5 - **Upon providing an acceptable description of the prestress monitoring and trending activities, the acceptance criteria, and corrective actions when acceptance criteria are not met, the applicant needs to include an adequate summary description of this information in the SAR Supplement consistent with 10CFR54.21(d).**

  See Sections 16.2.3.6 and 16.3.4 of the revised ANO-1 SAR Chapter 16 (attached).

- SAR Item 4.7.3 - **The applicant needs to provide a summary description of the monitoring, evaluation activities, optional corrective actions, and decision criteria for the aging of Boraflex in the spent fuel pool.**

  See Section 16.3.6 of the revised ANO-1 SAR Chapter 16 (attached).

Open Item 3.3.7.2.4-1 - Buried (inaccessible) medium-voltage cables, exposed to ground water typically do not have comparable accessible cables exposed to a similar environment that can serve as a sample for the inaccessible cables. For buried cables exposed to ground water that are within the scope of license renewal and subject to an aging management review, visual inspection is not sufficient for managing a reduced insulation resistance to ground, and potential electrical failure due to moisture intrusion, water treeing, and contamination so that the intended function will be maintained consistent with the applicant's current licensing basis for the period of extended operation in accordance with the requirements of 10CFR54.21(a)(3).

ANO-1 will implement either a testing or replacement program for the medium-voltage cables exposed to ground water. If a testing program is implemented, inaccessible medium-voltage cables exposed to significant moisture and voltage will be tested for the presence of aging effects. The specific type of test will provide an indication of cable insulation integrity and will be implemented prior to year 40. In addition, water collection in manholes containing in-scope, medium-voltage cables will be monitored and managed to reduce the cables' exposure to significant moisture. Significant moisture exposure is defined as periodic exposure to moisture that lasts more than a few days. Periodic exposure to moisture that lasts less than a few days (i.e., normal rainfall) is not significant. Significant voltage exposure is defined as being subjected to system voltage for more than twenty-five percent of the time. Acceptance criteria will be defined for the specific type of test to be performed and the specific cable to be tested. Appropriate corrective actions will be implemented when acceptance criteria are not met.

If periodic replacement is determined to be the most effective action, Entergy Operations will replace the medium-voltage cables in the scope of license renewal that are exposed to ground water at a periodic frequency. The frequency will be based on both site specific and industry operating experience.

The implementation of this testing or periodic replacement program will occur before the end of the current license period. Either the testing or periodic replacement options will provide an aging management program to manage the aging effects for these cables and ensure the required functions of the cables are maintained throughout the license renewal period.

<u>Open Item 4.5.3-1</u> - In response to an NRC Staff RAI, the applicant does not adequately describe the aging management program. Specifically, the applicant needs to provide additional information regarding the prestress monitoring and trending activities, the acceptance criteria, and corrective actions when acceptance criteria are not met. The applicant is requested to provide this information with respect to the prestress forces of the ANO-1 reactor building.

The following discussion of the attributes of the IWL Inspection Program provides the requested additional information.

<u>Prestress monitoring and trending:</u>
The tendon surveillance is conducted every five years as required by ASME Section XI, Subsection IWL. Trending is accomplished as required by 10CFR50.55a(b)(2)(ix)(B). The requirements for tendon surveillance and tendon force graphs for ANO-1 are included within the site tendon surveillance program procedures.

The IWL Inspection Program provides for the random selection of tendons. The surveillance of the selected tendons includes inspection of the tendon components, including taking wire and grease samples, inspection of the concrete around the tendon anchorage, and determining residual tendon force.

During the surveillance, lift-off forces for the tendons are measured and evaluated for adequacy as required by IWL. Graphs for each group of tendons (hoop, dome, and vertical tendons) provide the age related expected normalized tendon force plotted on a log-normal graph. These graphs are developed based on the tendon group and the aging effects on the reactor building concrete properties, the wire properties, and the initial prestress force. The lift off values obtained during the tendon surveillances are plotted on the graphs and trended to determine if the tendon system is performing as expected.

<u>Acceptance Criteria:</u>
The acceptance criteria are included in the site procedures for the reactor building tendon surveillance and concrete inspection. The tendon force graphs are compared with the actual forces found during the surveillance to determine if the residual prestress in the reactor building meets the minimum required prestress.

The minimum required tendon force for each of the tendon groups are: 1233 kips for the hoop tendons, 1274 kips for the vertical tendons, and 1252 kips for the dome tendons. Corrective actions will be taken should the projected tendon forces for a tendon group fall below the minimum required value before the next scheduled tendon surveillance.

<u>Corrective actions:</u>
Conditions that do not meet the acceptance criteria in the site procedures are documented in the site condition reporting system and per ANO-1 Technical Specification 6.12.4.1. Evaluations are performed and acceptability is determined. Corrective actions that are needed are tracked to completion through the condition reporting system.

Should trending indicate that prestress in a tendon group may be inadequate to meet the minimum required prestress before the next scheduled tendon surveillance, action will be taken to correct the problem. This may include re-tensioning, replacing tendons, or reanalysis of the reactor building to assure adequate prestress to meet design requirements.

<u>Open Item 4.7.3-1</u> - The applicant needs to provide the basis upon which the Staff can conclude that there is reasonable assurance that the effects of aging of Boraflex will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation, in accordance with 10CFR54.21(c)(1).

The analysis of Boraflex in the spent fuel storage racks is a time limited aging analysis. The analysis is not valid through the license renewal period and cannot be acceptably projected to the end of the license renewal period as documented in correspondence dated September 6, 2000 (1CAN090002). In accordance with 10CFR54.21(c)(1)(iii), the Boraflex Monitoring Program will provide reasonable assurance that the effects of aging on the intended function will be adequately managed for the period of extended operation. By letter dated September 6, 2000 (1CAN090002), Entergy Operations provided a response to RAI 4.7-1 concerning Boraflex monitoring. In order to address this open item, Entergy Operations is providing additional information in response to items (a) through (e) of RAI 4.7-1.

The LRA, Section 4.7, describes the TLAA for the degradation of Boraflex, which is currently used in Region I of the ANO-1 spent fuel storage racks as a neutron absorber. In response to Generic Letter (GL) 96-04, you committed to continued monitoring and analysis of the Boraflex degradation at ANO-1. The LRA, Section 4.7, states that the existing coupon monitoring program will be continued, as required, into the extended license period. In addition, monitoring of the spent fuel pool silica levels and perform silica evaluations will also be continued into the period of extended operation. These evaluations are based on the EPRI RACKLIFE system or its equivalent. Projected Boraflex performance will be assessed to confirm that a 5% subcriticality margin will be maintained as required.

Your response to GL 96-04 states that long-term and accelerated test location coupon specimens are periodically removed and inspected and that "the inspections provide an indication of the general condition of the Boraflex, including gross or unusual degradation." Long-term coupons are tested approximately every five years, while accelerated coupons are tested after each refueling. In addition, monitoring of the spent fuel pool silica levels, silica evaluations based on the EPRI RACKLIFE system or its equivalent, and assessment of the projected Boraflex performance to confirm a 5% subcriticality margin will continue through the next

evaluation period. These assessments will be performed each cycle prior to fuel receipt.

In order to complete the evaluation of this TLAA, the staff requests the following information:

(a) Clarify that the frequency of the inspection and testing as discussed above will be the same for the extended license period.

In the ANO response to Generic Letter 96-04 (0CAN109605), Entergy Operations committed to continued monitoring and analysis of Boraflex degradation at ANO-1. Entergy Operations will continue the coupon monitoring program into the extended license period. Although the accelerated coupons have been used and are no longer available as discussed in item (b) below, the long-term coupons are tested once every 5 years. These coupons have provided indications of Boraflex degradation. Entergy Operations will continue to monitor spent fuel pool silica levels and perform silica evaluations once per cycle. These evaluations are based on the EPRI RACKLIFE system. Boraflex performance will be projected to confirm the 5% subcriticality margin will be maintained as required.

(b) Are there sufficient long-term and accelerated coupons to continue the existing monitoring program through the end of the extended license period? If not, by what other means will indications of actual Boraflex degradation be obtained?

There are a sufficient number of long-term coupons to continue the existing program through the period of extended operation. The portion of the program for which the accelerated coupons were designed is complete.

(c) Describe the physical conditions that are observed during the inspection of the sampling coupons. Do they include inspections for discoloration, hardness and reduction of thickness? If not, what conditions are observed that are directly related to the degradation of the Boraflex?

ANO-1 currently has a procedure in place for examining and testing the spent fuel pool Boraflex test coupons. The coupon inspections consist of taking thickness measurements, density determinations, general visual inspection, and hardness testing. Neutron attenuation testing is also performed on the sampled Boraflex coupons. This testing more accurately determines areal Boron density.

(d) Boraflex panel degradation can be characterized by gap formation and a decrease in areal boron density. Clarify how these parameters are monitored by the ANO-1 program. If not, provide the technical bases for not monitoring these parameters.

The minimum as-designed Boraflex dimensions and the minimum designed areal Boron 10 densities with an assumed degradation of 10% are used in the ANO-1 criticality analysis. The ANO-1 criticality analysis assumes all the shrinkage is on the ends, which is more conservative than gap formation assumptions for the ANO-1 rack geometry. The Boraflex panels are assumed to shrink 4.1% in width. The Boraflex gap and shrinkage values are consistent with the EPRI Boraflex shrinkage model and gap measurements. Current RACKLIFE analysis indicates that there is less than 10% boron degradation.

These assumptions and analytical calculations have been correlated to industry data obtained through in-situ testing of a similar rack design to ANO-1. The results from the tested racks are conservatively applied to the ANO-1 racks based upon the tested racks having been subjected to higher doses, the spent fuel pool silica levels exceeding the concentrations seen at ANO-1, and the tested racks have a higher peak panel degradation. The tested racks have also been in service longer than the ANO-1 racks.

(e) Provide the results of current trending analyses that have been obtained by use of the RACKLIFE code. Do these results demonstrate that the 5% subcriticality margin of the spent fuel racks will be maintained for the extended period of operation? If not, describe the corrective actions that will be implemented to ensure that the 5% subcriticality margin will be maintained through the extended period of operation.

The results of the current Boraflex trending analysis demonstrate that the 5% subcriticality margin is being maintained; however, it will not be maintained for the period of extended operation. As previously discussed, this condition has been documented in accordance with the onsite Appendix B corrective action program. Corrective actions will be implemented to ensure that the 5% subcriticality margin will be maintained through the period of extended operation. Corrective actions may include modification of the spent fuel racks to incorporate a different neutron absorber material. Entergy Operations is committed to resolving this issue as documented in correspondence dated September 6, 2000 (1CAN090002).

Attachment B
Appendix A, SAR Supplement (Chapter 16) Changes

16.0 Aging Management Programs and Activities

Added text from RAI 3.3.1.4.4-2 (1CAN090004) in response to NRC SER Open Item 3.3-1, SAR Item 3.3.1.2.3. This information is about the corrective action program and the quality assurance program.

16.1.2 Electrical Component Inspection

Added text in response to NRC SER Open Item 3.3-1, SAR Item 3.3.7.4.

16.1.3 Heat Exchanger Monitoring

Minor changes to resolve differences between ANO-1 LRA Appendix B and Appendix A. Clarified scope of components for which this program is credited.

16.1.4 Pressurizer Examinations

Made the word "weld" plural.

16.1.4.1 Pressurizer Cladding Examination

For clarification added additional detail from program description in Appendix B.

16.1.4.2 Pressurizer Heater Bundle Penetration Welds Examination

Added additional detail from ANO-1 LRA Appendix B describing the option of using results from Oconee inspection.

16.1.7 Wall Thinning Inspection

Added clarification to better describe the components covered by the program. Also added the timing of the program as stated in the ANO-1 LRA Appendix B.

16.2.2 Alternate AC Diesel Generator Testing and Inspections

Changed "or" to "and" in three places to clarify the applicable aging effects.

16.2.3.6 IWL Inspections

Added text in response to NRC SER Open Item 3.3-1, SAR Item 4.5.5. Text is from SER Open Item 4.5.3-1 response.

16.2.3.7 Augmented Inspections

Added sentence regarding inspections. The RCP visual inspection is the same inspection discussed in the ANO-1 LRA under IWB Inspections. The sodium hydroxide portion of this change is due to RAI 3.3.3.3-14 response (1CAN090004), and is in response to NRC SER Open Item 3.3-1, SAR Item 3.3.3.3. The chilled water stainless steel tubing portion of this change credits an additional program for managing aging effects on the chilled water system.

16.2.5 Boric Acid Corrosion Prevention

Changed "loss of pressure integrity" to "loss of mechanical closure integrity" to agree with terminology used throughout the ANO-1 LRA.

16.2.6.2 Secondary Chemistry Monitoring

Added "emergency feedwater system and main steam system" to "Scope" per RAI 3.3.5-11(a) response (1CAN090004).

16.2.6.4 Diesel Fuel Monitoring

Added sentence to explicitly state the aging effects being managed by this program.

16.2.6.5 Service Water Chemical Control

Deleted "auxiliary cooling water system" from the discussion of systems that credit this program. The mechanical components of the auxiliary cooling water system are not subject to aging management review. This is consistent with the ANO-1 LRA.

16.2.7 Control Rod Drive Mechanism Nozzle and Other Vessel Closure Penetration Inspection Program

Revised in response to NRC SER Open Item 3.3-1, SAR Item 3.3.2.4.3. It is consistent with the "Acceptance Criteria" of the program description in Appendix B of the ANO-1 LRA.

16.2.8.3 Fire Suppression Water Supply System Surveillance

Added "loss of mechanical closure integrity" to the discussion of "Aging Effects" per RAI 3.3.4.3.2.2-2(a) response (1CAN090004). Listed fouling of heat exchangers and loss of material from external surfaces to be consistent with ANO-1 LRA Table 3.4-2 and the response to RAI 3.3.4.3.1-2(d) (1CAN090004).

16.2.8.4 Fire Suppression Sprinkler System Surveillance

Deleted "fouling" from the "Aging Effects" discussion. This is consistent with ANO-1 LRA Table 3.4-2.

16.2.8.5 Fire Water Piping Thickness Evaluation

Deleted "stainless steel" from the "Aging Effects" discussion to agree with RAI 3.3.4.3.2.2-3(a) response (1CAN090004).

16.2.8.6 Control Room Halon Fire System Inspection

Added "halon discharge nozzles" to the "Scope" discussion to agree with ANO-1 LRA Table 3.4-6

16.2.9 Flow Accelerated Corrosion Prevention

Changed discussion to clarify distinction between aging effect and aging mechanism.

16.2.13 Maintenance Rule

Added "loss of mechanical closure integrity" as an aging effect. Reference RAI .3.4.3.2-9(b) response (1CAN090004). Added last sentence in response to SER Open Item 3.3-1, SAR Item 3.3.1.3.3.

16.2.15 Preventive Maintenance

Added "cracking" of the expansion joints as another aging effect managed by the subject preventative maintenance tasks. Added two preventative maintenance activities per RAI 3.3.4.3.2.2-2(c) and 3.3.4.3.2-10 responses (1CAN090004).

16.2.16 Reactor Building Leak Rate Testing

Deleted "changes in material properties" from the "aging effects" discussion to agree with ANO-1 LRA Section 3.0 tables.

16.2.17 Reactor Building Sump Closeout Inspection

Deleted the phrase, "due to the presence of borated water" from the "Aging Effects" discussion since the intent was to include the aging effect regardless of potential causes.

16.2.19 Service Water Integrity

Added the item regarding ECP return line epoxy coating to be consistent with the discussion under "Method" in Appendix B of the ANO-1 LRA.

16.2.21.5 Emergency Diesel Generator Testing and Inspections

Added the last sentence to address RAI 3.3.4.3.2.7(b) response (1CAN090004).

16.3.2 Metal Fatigue

Revised TLAA discussion to address SER Open Item 3.3-1, SAR Item 4.3.4. The revision is consistent with the response to RAI 4.3.3-2 (1CAN090002).

16.3.4 Concrete Reactor Building Tendon Prestress

Revised TLAA discussion to address SER Open Item 3.3-1, SAR Item 4.5.5 and SER Open Item 4.5.3-1.

16.3.6 Aging of Boraflex in Spent Fuel Pool Racks

Revised in response to SER Open Item 3.3-1, SAR Item 4.7.3. Added reference to response to Generic Letter 96-04. The existing discussion is consistent with the response provided to SER Open Item 4.7.3-1.

16.3.9 Leak-Before-Break

Revised to include reference to FTI report that evaluated acceptability through period of extended operation.

Appendix A

Safety Analysis Report Supplement

INTRODUCTION

This appendix contains the SAR Supplement required by 10CFR54.21(d) for the ANO-1 License Renewal Application. The LRA contains the technical information required by 10CFR54.21(a) and 10CFR54.21(c). Section 3.0 and Appendix B of the ANO-1 LRA provide descriptions of the programs and activities that manage the effects of aging for the period of extended operation. Section 4.0 contains the evaluations of the time-limited aging analyses for the period of extended operation. These sections have been used to prepare the program and activity descriptions that are contained in the SAR Supplement. The SAR Supplement will be incorporated into the ANO-1 SAR following issuance of the renewed operating license for ANO-1. Upon inclusion of the SAR Supplement in the ANO-1 SAR, changes to the descriptions of the programs and activities will be made in accordance with 10CFR50.59.

CHAPTER 4 CHANGES

4.1.2.6 Service Lifetime

The original design service lifetime for the major RCS components ~~is~~ was 40 years. The number of cyclic system temperature, pressure, and operational changes (Table 4-8) ~~is~~ was based on operation for this design lifetime. The commencement date for the original design service life was the date of the Construction Permit which approved the PSAR for Unit 1, which is December 6, 1968. However, in 1990, per License Amendment 131, an extension was granted to allow the operating license term to be changed to start at the issuance of the operating license to allow a 40-year service life that does not include the construction time period and end on May 20, 2014. A new operating license has been granted to extend the licensed term an additional 20 years to May 20, 2034. This was justified based on design transient cycles. The reactor coolant system was originally qualified using a conservative estimate of design cycles for a 40 year life. The design life is not dependent on years of service. The design life is dependent on fatigue cycles. In evaluations performed by the NSSS vendor, the actual cycles were extrapolated to 60 years. For the major RCS components, the design cycles exceeded the estimated cycles for a 60-year life. The actual transient cycles are tracked and documented to ensure they are maintained below the allowable number of design cycles as further discussed in Section 16.2.21. Table 4-8 shows the complete listing of transients used in the design of components within the Reactor Coolant Pressure Boundary. Records of significant transients are available from the ~~daily~~ periodic reviews of the Shift Superintendent's log. A record of all significant transients is maintained by the technical support staff.

4.1.2.8 Vessel Radiation Exposure

The reactor vessel is the only RCS component exposed to a significant level of neutron irradiation and is therefore the only component subject to material radiation damage. The maximum exposure from fast neutrons (E > 1.0 MeV) has been computed to be less than 3.0×10^{19} neut/cm² over a 40-year life with an 80 percent load factor. Revised calculations, based on results of the Reactor Vessel Material Surveillance Program and reduced leakage fuel cycle configurations, indicate that the maximum exposures will be less than half of this value. The maximum inside surface fast neutron fluence at 48 EFPY is projected to be 1.44×10^{19} neut/cm². Reactor vessel irradiation calculations are described in Section 4.3.3.

4.3.3 REACTOR VESSEL

(Excerpted from Nil Ductility Transition Temperature (NDTT))

Revised calculations, based on results of the Reactor Vessel Material Surveillance Program and reduced leakage fuel cycle configurations, indicate that the maximum fluence value will be less than half of the originally estimated $3x10^{19}n/cm^2$. The maximum inside surface fast neutron fluence at 48 EFPY is projected to be $1.44x10^{19}$ n/cm^2. The corresponding EOL transition temperature is similarly reduced while the minimum upper shelf energy value is increased.

(Excerpted from Flux and Total Integrated Flux (nvt) at Reactor Vessel Wall)

Revised calculations, based on results of the Reactor Vessel Material Surveillance Program and reduced leakage fuel cycle configurations, indicate that the maximum fluence values will be less than the originally calculated values. The revised fluence value for 40 years at 80 percent load is 1.10×10^{19} n/cm^2. This value was used to respond to the Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock, 10CFR50.61. The projected fast fluence value has since been revised lower to 8.71×10^{18} n/cm^2, as reported to the NRC via 1CAN119608. The projected maximum fast fluence value for 48 EFPY has been determined to be 1.44×10^{19} n/cm^2.

(Excerpted from Expected NDTT Shift)

Revised values of calculated vessel exposure have been used to determine the increase in the NDTT in accordance with 10CFR50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock." The results have been reported to and accepted by the NRC (1CNA128603). For the 40-year exposure at 80 percent load, the calculated value of the NDTT is 264°F. The calculation and supporting documentation are described in BAW-1895 (Jan. 1986). "Pressurized Thermal Shock Evaluation in Accordance with 10CFR50.61 for B&W Owners Group Reactor Pressure Vessels." Since the submittal of BAW-1895, the fourth capsule report has been produced and submitted. This capsule report indicates the NDTT value is 257°F based on a further reduction in the projected fluence value.

BAW-2251-A shows the RT_{PTS} to be 278°F for 48 EFPY for a circumferential weld.

The NDTT shift is factored into the plant startup and shutdown procedures so that full operating pressure is not attained until the reactor vessel temperature is about DTT. The heatup and cooldown curves are given in Technical Specification 3.1.2, "Pressurization, Heatup and Cooldown Limitations. The total stress in the vessel wall due to both pressure and the associated heatup and cooldown transient is restricted to 5,000 - 10,000 psi, which is below the threshold of concern for safe operation. An adjusted 100°F per hour heatup rate can be maintained throughout life. An adjusted rate is one in which the

pressure is held constant to maintain stresses at the desired low level while temperatures are at a level below DTT. A 100°F per hour temperature increase is maintained until DTT is passed and pressure can be raised to a new higher level. These operating restrictions are based on the NRL generalized fracture analysis diagram, which is a semi-empirical method of material selection and approximate analysis to prevent brittle fracture. This diagram plots failure stress (normalized to yield) as a function of temperature referenced to the NDTT for a family of finite flaw sizes. The parametric crack size curves were determined partially by fracture mechanics and partially by plotting actual failure data. The assumed flaw for this analysis was slightly greater than 24 inches.

CHAPTER 5 CHANGES

5.2.1.4.7.3 Loads

All load combinations are considered in the above analysis. The following fatigue loads are considered in liner design:

A. Thermal cycling due to annual outdoor temperature variations - Daily temperature variations will not penetrate a significant distance into the concrete shell to appreciably change the average temperature of the shell relative to the liner plate. The number of cycles for this loading is ~~40~~ 60 cycles for the plant life of ~~40~~ 60 years.

B. Thermal cycling due to reactor building interior temperature varying during the heatup and cooldown of the reactor system - The number of cycles for this loading is assumed to be 500.

C. One cycle was assumed for thermal cycling due to a DBA.

CHAPTER 6 CHANGES

6 ENGINEERED SAFEGUARDS

(Excerpted from Chapter 6)

Operability of engineered safeguards equipment is assured in several ways. Much of the equipment in these systems function during normal reactor operation thus providing a constant check on operational status. Where equipment is used for emergency functions only, such as in the Reactor Building Spray System, the systems have been designed to permit meaningful periodic tests. Operational reliability has been achieved by using proven component designs wherever possible and/or by conducting tests. Quality control and assurance requirements are implemented during the design, manufacture, and installation of the engineered safeguards components and systems to assure that a high quality level is maintained. The quality program is based upon the use of accepted industry codes and standards as well as supplementary test and inspections. The resultant high quality level of the components gives assurance that they will perform their intended function under the worst anticipated conditions following a LOCA. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of a Design Basis Accident (DBA). All equipment must ~~operate for the designed 40-year~~ remain functional throughout the life of the plant. Certain safety-related equipment must operate during the design plant life as well as function as required during and following a DBA at the end of plant life.

CHAPTER 11 CHANGES

11.2.1.2 Radiation Exposure of Materials and Components

~~No regulations similar to those established for the protection of individuals exist for materials and components. However, m~~Materials and components are selected on the basis that their design radiation exposure will not cause significant changes in their physical properties which adversely affect operation of equipment during the design life of the plant. Materials for equipment required to operate under accident conditions are selected on the basis of the additional exposure received in the event of a DBA. The approximate radiation damage threshold for various materials is shown in Table 11-13.

NEW CHAPTER 16

16.0 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The integrated plant assessment for license renewal identified several new programs and activities, modifications to existing programs, and existing programs, necessary to continue operation of ANO-1 during the additional 20 years beyond the initial license term. This chapter describes these programs and activities. The ANO-1 corrective action program, which includes the confirmation process to assure that the cause of the condition is determined and corrective action taken to preclude repetition, and the administrative (document) control program, which governs site procedures, were credited for license renewal. These programs are in accordance with the corporate quality assurance program pursuant to 10CFR Part 50, Appendix B. These programs apply to all of the programs credited for license renewal.

16.1 NEW ACTIVITIES

16.1.1 BURIED PIPE INSPECTION

Buried Pipe Inspections will be performed to ensure that a loss of material due to external surface corrosion of buried piping is adequately managed. The safety-related portions of underground carbon steel piping on the service water and fuel oil systems are within the scope of this inspection. The aging effect addressed by the Buried Pipe Inspection is a loss of material due to corrosion of the external surfaces of pipe caused by loss of the protective coating. This inspection will be initiated prior to the end of the initial 40-year license term.

16.1.2 ELECTRICAL COMPONENT INSPECTION

The Electrical Component Inspection Program will inspect splices, connectors, and cables within the scope of license renewal that are located in areas that may be conducive to accelerated aging. The scope of the inspection program includes cables exposed to elevated temperatures, wet environments, or corrosive chemicals. The scope also includes cables that can experience elevated temperatures due to the current they are carrying, connectors used in impedance-sensitive circuits, and cable splices subject to aging-related stressors. The aging effect for cables and cable splices is a change of material properties, as evidenced by cracking or discoloration of the insulation or by degradation of a tested parameter. The aging effect for connectors in impedance-sensitive circuits is a change of material due to corrosion of connector pins. The Electrical Component Inspection Program will be formally implemented and the first inspection or test of in-scope cables, splices, and connectors will be completed prior to the expiration of the initial 40-year licensing term. Inaccessible medium-voltage cables exposed to significant moisture and voltage will either be tested for the presence of aging effects or a replacement program

for these cables will be developed. If a periodic replacement of medium-voltage underground cables is determined to be the most effective action for this type of cable, ANO-1 will define the frequency for replacement prior to the expiration of the initial 40-year licensing term. The frequency will be based on site specific and industry operating experience.

16.1.3 HEAT EXCHANGER MONITORING

The Heat Exchanger Monitoring Program will inspect heat exchangers to the extent required to ensure seismic qualification is maintained. The Heat Exchanger Monitoring Program manages aging effects on the following safety-related systems and components: reactor building coolers, emergency diesel generator jacket cooling water heat exchangers, make-up pump lube oil coolers, make-up pump room coolers, decay heat room coolers, decay heat system heat exchangers, electrical room chillers and coolers, control room chillers and coolers, emergency feedwater system heat exchangers, and chilled water condensers and evaporators. The aging effects addressed by the Heat Exchanger Monitoring Program are cracking and loss of material that could result in degradation in the seismic qualification of the heat exchangers. Inspection will be initiated prior to the end of the initial 40-year license term.

16.1.4 PRESSURIZER EXAMINATIONS

The Pressurizer Examinations include two specific examinations: the pressurizer cladding and the pressurizer heater penetration welds examination.

16.1.4.1 Pressurizer Cladding Examination

The pressurizer cladding examination will assess the condition of the pressurizer cladding. The scope of this activity will include the cladding and attachment welds to the cladding of the pressurizer. The aging effect is cracking of cladding by thermal fatigue, which may propagate to the underlying ferritic steel. The volumetric examinations of the pressurizer items that are most susceptible to thermal fatigue will provide assurance that cracking of cladding has not extended into the base metal of the pressurizer. These examinations are included in the ISI program and will be carried forward to the period of extended operation.

16.1.4.2 Pressurizer Heater Bundle Penetration Welds Examination

The pressurizer heater bundle penetration welds examination will be completed at ANO-1 or on the Oconee pressurizer heaters and will assess the condition of the pressurizer heater penetration welds. This examination will be applicable to the heater sheath-to-diaphragm plate penetration welds inside the pressurizer. The aging effect is cracking at the heater bundle penetration welds, which may lead to reactor coolant leakage. The heater bundle

examination may occur prior to entering the period of extended operation or during the period of extended operation.

16.1.5 REACTOR VESSEL INTERNALS AGING MANAGEMENT

Ongoing industry efforts are aimed at characterizing the aging effects requiring management associated with the reactor vessel internals. Further understanding of these aging effects will be developed by the industry over time and will provide additional bases for the inspections under this program. Entergy Operations will participate in the BWOG Reactor Vessel Internals Aging Management Program and other industry programs, as appropriate, to continue investigation of aging effects requiring management for the reactor vessel internals. These activities will assist in establishing appropriate monitoring and inspection programs for the reactor vessel internals. Entergy Operations will provide periodic updates after the completion of significant milestones in the preparation of the Reactor Vessel Internals Inspection, commencing within one year of the issuance of the renewed license. Entergy Operations will submit a report to the NRC, at or about, the end of the initial 40-year operating license term. This report will summarize the current understanding of aging effects applicable to the reactor vessel internals and will contain the Entergy Operations' inspection plan, including methods for each inspection. Entergy Operations will perform the Reactor Vessel Internals Inspection. Should data or evaluations indicate that this inspection can be modified or eliminated, Entergy Operations will provide plant-specific justification to demonstrate the basis for the modification or elimination. The purpose of the Reactor Vessel Internals Inspection is to inspect and examine the reactor vessel internals to assure the aging effects will not result in loss of the intended functions of the internals during the period of extended operation. The inspection applies to the reactor vessel internals. This inspection will begin during the period of extended operation. The aging effects for the reactor vessel internals include cracking due to either stress corrosion or irradiation assisted stress corrosion, reduction of fracture toughness due to irradiation embrittlement, dimensional changes due to void swelling, and loss of bolted closure integrity due to stress relaxation.

16.1.6 SPENT FUEL POOL MONITORING

The Spent Fuel Pool Monitoring Program will manage the aging effects requiring management of the ANO-1 spent fuel pool liner. Stress corrosion cracking is possible from the external surface of the liner in weld heat-affected zones since this was not verified to be chloride-free during construction. This program will be initiated before the end of the initial 40-year license term.

16.1.7 WALL THINNING INSPECTION

Wall Thinning Inspections will be performed to ensure wall thickness is above the minimum required to avoid leaks or failures under normal conditions and postulated transient and accident conditions, including seismic events. Wall Thinning Inspections will cover the following safety-related systems and components: EFW pump casing and

carbon steel discharge piping and valves, EFW steam supply components downstream of steam admission valves, EFW steam exhaust piping and valves, EFW carbon steel cooling water, seal water, and instrument piping and valves, EFW turbine lube oil cooler, carbon steel EFW supply header piping and valves (condensate supply), NaOH tank, carbon steel piping and components of the main steam system, and carbon steel components of penetrations 11, 42, 43, 48, 49, 51, 52, 54, 58, 59, 60, 62, and 64. The aging effect to be addressed by Wall Thinning Inspection is a loss of material due to corrosion of the internal surfaces of carbon steel piping and components. This program will be initiated before the end of the initial 40-year license term.

16.2 EXISTING ACTIVITIES

16.2.1 ALLOY 600 AGING MANAGEMENT

The Alloy 600 Aging Management Program will manage cracking by PWSCC of Alloy 600 and Alloy 82/182 locations for the period of extended operation. The Alloy 600 Aging Management Program will be applicable to the Alloy 600 items and Alloy 82/182 weld material in the RCS, including the hot leg flow meter element. The aging effect managed by the Alloy 600 Aging Management Program is cracking of Alloy 600 items and Alloy 82/182 weld material in the RCS.

16.2.2 ALTERNATE AC DIESEL GENERATOR TESTING AND INSPECTIONS

The Alternate AC Diesel Generator Testing and Inspections ensures that the effects of aging are managed before the loss of the intended functions of the system. The Alternate AC Diesel Generator Testing and Inspections applies to the station blackout diesel and its components. The aging effects addressed by the Alternate AC Diesel Generator Testing and Inspections include: loss of material and loss of mechanical closure integrity for the starting air subsystem components; loss of material and loss of mechanical closure integrity for the intake combustion air subsystem components; loss of material, fouling, and loss of mechanical closure integrity for the intake air aftercooler; loss of material for carbon steel components, cracking of the stainless steel components and loss of mechanical closure integrity for the exhaust subsystem components; loss of mechanical closure integrity for the lube oil subsystem components; fouling, loss of material from wear, and loss of mechanical closure integrity for the lube oil cooler; loss of material and loss of mechanical closure integrity for the cooling water subsystem components; fouling and a loss of material for the AAC radiator; loss of material from wetted portions of the exhaust fan housings; and fouling of the fuel oil heat exchanger.

16.2.3 ASME SECTION XI INSERVICE INSPECTION

16.2.3.1 IWB Inspections

The ASME Section XI, Subsection IWB Inspections under the scope of the Inservice Inspection Plan identifies and corrects degradation of ASME Class 1 pressure retaining components and their integral attachments. The scope of the ASME Section XI, Subsection IWB Inspections, credited for license renewal, is identified specifically for each component and for applicable component features in the ISI Plan. The aging effects managed as part of the ASME Section XI, Subsection IWB Inspections include cracking, loss of mechanical closure integrity at bolted connections, and loss of material. In addition, a one-time visual inspection of a reactor coolant pump casing will be performed prior to the end of the initial 40-year license term.

16.2.3.2 IWC Inspections

ASME Section XI, Subsection IWC Inspections under the scope of the ANO-1 Inservice Inspection Plan identify and correct degradation of ASME Class 2 pressure retaining components and their integral attachments. The scope of the ASME Section XI, Subsection IWC Inspections, credited for license renewal, includes components on the following systems: core flood, RBS, main feedwater, spent fuel, service water, HPI/makeup and purification, LPI/decay heat, EFW, main steam, reactor building isolation, and chilled water system. The aging effects managed as part of the ASME Section XI, Subsection IWC Inspections include cracking, loss of mechanical closure integrity, and loss of material.

16.2.3.3 IWD Inspections

ASME Section XI, Subsection IWD Inspections under the scope of the Inservice Inspection Plan identify and correct degradation of ASME Class 3 pressure-retaining components. The scope of the ASME Section XI, Subsection IWD Inspections, credited for license renewal, includes components on the following systems: service water, spent fuel, main steam, EFW, sodium hydroxide, and condensate storage. The aging effects managed as part of the ASME Section XI, Subsection IWD Inspections include cracking, loss of mechanical closure integrity, and loss of material.

16.2.3.4 IWE Inspections

ASME Section XI, Subsection IWE Inspections under the scope of the Inservice Inspection Plan identify and correct degradation of Class MC pressure retaining components, their integral attachments, the metallic shell and penetration liners of Class CC pressure retaining components, and their integral attachments. The scope of the ASME Section XI, Subsection IWE Inspections, credited for license renewal includes

inspections of the reactor building liner plate. The aging effect managed as part of the ASME Section XI, Subsection IWE Inspections is a loss of material of the steel surfaces.

16.2.3.5 IWF Inspections

ASME Section XI Inservice Inspection Program, IWF Inspections identify and correct degradation of ASME Class 1, 2, 3, or MC component supports. The aging effects managed as part of the ASME Section XI, Subsection IWF include cracking, loss of material, and change in material properties.

16.2.3.6 IWL Inspections

ASME Section XI Inservice Inspection Program, IWL Inspections provides instructions and documentation requirements for assessing the quality and structural performance of the reactor building's post-tensioning system and concrete components. The scope includes the reactor building's post-tensioning system and concrete components. The aging effects are loss of material for tendon wires and anchorage and cracking and change in material properties for prestressed concrete components. The IWL inspections and associated trending are performed in accordance with 10CFR50.55a(b)(2)(ix)(B). Acceptance criteria are in accordance with ASME Code, Section XI, Subsection IWL requirements. Corrective actions may include re-tensioning, replacing tendons, or reanalysis of the reactor building to assure adequate tendon prestress to meet design requirements.

16.2.3.7 Augmented Inspections

The ASME Section XI, Augmented Inspections identify and correct degradation of components outside of the jurisdiction of ASME Section XI. Augmented periodic inspections are completed for several main feedwater and main steam system welds, not in the Class 2 piping, to support the high energy line break analysis. Augmented inspections are completed for the BWST header including the lines from the reactor building sump. Augmented inspections that will be added to the program because of license renewal include a special augmented inspection on the welds of the piping wetted by the reactor building sump water, some supplemental inspections of the "Q" stainless piping of the main steam system, at least a one-time inspection of the penetration 68 piping and components and the decay heat pump room drain valves, and special inspections of penetrations 10, 47, 58, and 64. Other augmented inspections include a visual inspection of pressure retaining surfaces of one RCP, volumetric/non-destructive inspection of sodium hydroxide system stainless steel piping and valves, and volumetric/non-destructive inspection of chilled water system stainless steel tubing and valves. The aging effects managed by these inspections are cracking and loss of material. The new inspections will be initiated prior to the end of the initial 40-year license term.

16.2.3.8 <u>Small Bore Piping and Small Bore Nozzles Inspections</u>

The Small Bore Piping and Small Bore Nozzles Inspections identify aging effects on small bore piping and nozzles. The small bore piping and small bore nozzles, within the scope of this program, are RCS piping and nozzles less than 4-inch NPS that do not receive volumetric inspection in accordance with ASME Section XI. The aging effect managed by this program is cracking. A risk-informed ISI method has been implemented to select RCS piping welds for inspection. The risk-informed approach consists of two essential elements. A degradation mechanism evaluation is performed to assess the failure potential of the piping system under consideration, and a consequence evaluation is performed to assess the impact on plant safety in the event of a piping failure. The results from these two independent evaluations are coupled to determine the risk significance of piping segments within the system, and priority is then given to the most risk significant piping segments during the selection of RCS piping welds for inspection.

16.2.4 BOLTING AND TORQUING ACTIVITIES

Bolting and Torquing Activities prevent degradation of bolting or identify and correct degradation of bolting. The scope of Bolting and Torquing Activities applies to pressure boundary bolting applications associated with components within the scope of license renewal. Applications include bolted flange connects for vessels (i.e., manways and inspection ports), flanged joints in piping, body-to-body joints in valves, and pressure-retaining bolting associated with pumps or valves and miscellaneous process components. The aging effects addressed by Bolting and Torquing Activities are cracking, loss of material, and loss of mechanical closure integrity.

16.2.5 BORIC ACID CORROSION PREVENTION

The Boric Acid Corrosion Prevention Program prevents corrosion damage due to leakage from the borated water systems. The Boric Acid Corrosion Prevention Program is concerned with the RCS and other structures and components containing, or exposed to, borated water. This program is credited with monitoring the boric acid corrosion of carbon steel external surfaces of structures and components exposed to leakage from borated water. Carbon steel is utilized for bolting on many of the systems that contain borated water. This program manages the loss of material of bolts that could eventually result in a loss of mechanical closure integrity for bolted connections.

16.2.6 CHEMISTRY CONTROL

The following subsections address the individual ANO-specific chemistry control programs in more detail:

- Primary Chemistry Monitoring
- Secondary Chemistry Monitoring
- Auxiliary Systems Chemistry Monitoring

- Diesel Fuel Monitoring
- Service Water Chemical Control

16.2.6.1 Primary Chemistry Monitoring

The Primary Chemistry Monitoring Program maximizes long-term availability of primary systems by minimizing system corrosion, fuel corrosion, and radiation field build-up. The scope of the Primary Chemistry Monitoring Program includes sampling activities and analysis on the following systems: RCS, borated water storage tanks, spent fuel pool system, letdown purification demineralizers, and reactor makeup water. The Primary Chemistry Monitoring Program provides assurance that an elevated level of contaminants and oxygen does not exist in the systems covered by the program. This prevents or minimizes the occurrence of cracking and other aging effects.

16.2.6.2 Secondary Chemistry Monitoring

The Secondary Chemistry Monitoring Program maximizes the availability and operating life of major components. The scope of the Secondary Chemistry Monitoring Program includes sampling activities and analysis on the main feedwater system, emergency feedwater system, main steam system, condensate storage system, and steam generators. The aging reviews for many of the safety-related, non-Class 1 systems also indirectly credit the Secondary Chemistry Monitoring Program since the condensate storage tanks are used as a source of makeup water to these systems. The Secondary Water Chemistry Monitoring Program ensures the levels of contaminants and oxygen are maintained within a range that prevents or minimizes the occurrence of loss of material and other aging effects.

16.2.6.3 Auxiliary Systems Chemistry Monitoring

The Auxiliary Systems Chemistry Monitoring Program maximizes the availability and operating life of the components used for the closed cooling loops. The scope of the Auxiliary Systems Chemistry Monitoring Program is limited to sampling activities and analysis on the ICW system, chilled water systems, emergency diesel generators, and the AAC diesel generator. The Auxiliary Systems Chemistry Monitoring Program is credited with minimizing the loss of material due to corrosion, cracking, fouling, and loss of mechanical closure integrity.

16.2.6.4 Diesel Fuel Monitoring

The Diesel Fuel Monitoring Program ensures that adequate diesel fuel quality is maintained to prevent plugging of filters, fouling of injectors, and corrosion of the fuel systems. The scope of the Diesel Fuel Monitoring Program is limited to sampling activities and analysis on the following tanks: bulk fuel oil storage tank, EDG fuel tanks, EDG day tanks, fire pump diesel day tank, and the AAC diesel generator day tank. The

aging management reviews credit the sampling and monitoring as providing an adequate control of the fuel oil to ensure water and contamination (including microbiological) are not present in the system. The Diesel Fuel Monitoring Program is credited with minimizing fouling, cracking, and loss of material.

16.2.6.5 Service Water Chemical Control

The Service Water Chemical Control Program maximizes the availability and operating life of the components in the service water system. The scope of the Service Water Chemical Control Program includes sampling activities and analysis on the service water system. The scope also includes chemical injection into the service water bays. The fire protection system takes suction from the service water bays. The Service Water Chemical Control Program has been credited for aging management in the service water system and the fire protection system since these systems draw suction from the intake structure. The chemical additions are credited only with reducing corrosion, not eliminating this mechanism.

16.2.7 CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE PENETRATIONS INSPECTION

The CRDM Nozzle and Other Vessel Closure Penetrations Inspection Program verifies the assumptions made in the BWOG safety evaluation of the susceptibility and consequence of PWSCC in B&W-designed CRDM nozzles. The scope of the program includes the B&W-designed reactor vessel closure head CRDM nozzles and other closure head penetrations. The aging effect is PWSCC of Alloy-600 nozzles with partial penetration welds that cause high circumferential residual stresses on the inner diameter of the nozzles opposite the welds. If an ANO-specific inspection program is determined to be necessary, the axial flaws detected during inspection will be analyzed and evaluated using the NUMARC acceptance criteria. Circumferential flaws will be analyzed and addressed with the NRC on a case-by-case basis.

16.2.8 FIRE PROTECTION

Fire Protection Program activities, with respect to aging management, include: fire barrier inspections, fire hose station inspections, fire suppression water supply system surveillance, fire suppression sprinkler system surveillance, fire water piping thickness evaluation, control room halon fire system inspection, NFPA 25 testing of sprinkler head components that are 50 years old, and RCP oil collection system visual inspection.

16.2.8.1 Fire Barrier Inspections

Fire Barrier Inspections provide for periodic surveillance of fire barriers separating redundant safe shutdown systems to assure that they perform their separation functions. The scope includes 10CFR50.48-required fire walls and fire floors as indicated on the fire protection drawings. Fire doors/hatches, fire damper mountings, fire wraps, and

penetration fire stops associated with 10CFR50.48-required fire walls and fire floors are within the scope. The aging effects for fire barriers are cracking, loss of material, and change in material properties.

16.2.8.2 Fire Hose Station Inspections

The Fire Hose Station Inspections assure that manual fire suppression is available to safety-related equipment. Fire hose reels associated with 10CFR50.48-required fire hose stations are within the scope of license renewal. The aging effect for fire hose reels is a loss of material.

16.2.8.3 Fire Suppression Water Supply System Surveillance

This surveillance verifies operability of fire suppression water supply system components. The Fire Suppression Water Supply System Surveillance applies to fire water system supply piping and valves. The surveillance applies to several diesel fire pump subsystems including the intake air, exhaust, lube oil, and cooling water. Fire protection system heat exchangers are also within the scope of this surveillance. This program verifies that loss of material due to internal surface corrosion of carbon steel, stainless steel, brass, or bronze components is managed. This program is credited with managing fouling of heat exchangers. Cracking of stainless steel, brass, or bronze components is also an aging effect being managed. This program also manages the loss of mechanical closure integrity and the loss of material from external surfaces for fire protection components.

16.2.8.4 Fire Suppression Sprinkler System Surveillance

This surveillance verifies operability of fire suppression sprinkler system components. Within the scope of license renewal, the Fire Suppression Sprinkler System Surveillance applies to fire suppression sprinkler system piping, valves, and nozzles. This surveillance manages loss of material due to internal surface corrosion of carbon steel, stainless steel, brass or bronze components. Cracking of stainless steel, brass, or bronze components is also an aging effect being managed.

16.2.8.5 Fire Water Piping Thickness Evaluation

The Fire Water Piping Thickness Evaluation provides a method for the examination and evaluation of pipe wall thickness changes in the fire water system. Within the scope of license renewal, the Fire Water Piping Thickness Evaluation applies to fire water system piping. A loss of material by internal surface corrosion of cast iron or carbon steel fire water system components is the aging effect managed by the Fire Water Piping Thickness Evaluation.

16.2.8.6 Control Room Halon Fire System Inspection

The Control Room Halon Fire System Inspection assures that frequently manipulated components are free of aging effects. The components within the scope of the Control Room Halon Fire System Inspection are the halon discharge nozzles, the halon discharge tube assembly and the halon pilot header flexible tubing and fittings and discharge tube assembly fittings. The aging effects addressed by the Control Room Halon Fire System Inspection are loss of material due to wear from frequent manipulations and cracking.

16.2.8.7 Reactor Coolant Pump Oil Collection System Visual Inspection

The Reactor Coolant Pump Oil Collection System Inspection ensures integrity of the reactor coolant pump oil leakage collection system. The Reactor Coolant Pump Oil Collection System Inspection applies to the shrouds, drip pans, dammed areas, accessible piping, collection tanks, and spray protection. The aging effects addressed by the Reactor Coolant Pump Oil Collection System Inspection are a loss of material and a loss of mechanical closure integrity. These aging effects would be caused by general corrosion of the carbon steel internal surfaces or external surfaces due to the potential for water leakage into the system.

16.2.9 FLOW ACCELERATED CORROSION PREVENTION

The Flow Accelerated Corrosion Prevention Program provides a programmatic approach for identifying, inspecting, and managing loss of material for components that are adversely affected by flow accelerated corrosion (also known as erosion/corrosion). Within the scope of license renewal, only the main feedwater and main steam systems are identified as susceptible to flow-accelerated corrosion. The aging mechanism is flow-accelerated corrosion, a phenomenon that results in metal loss from components made of carbon steel, which occurs only under certain conditions of flow, chemistry, geometry, and material. The aging effect is loss of material. The aging management reviews credit this program with determining which systems are susceptible to flow-accelerated corrosion and monitoring the loss of material for those systems.

16.2.10 INSPECTION AND PREVENTIVE MAINTENANCE OF THE POLAR CRANE

The program provides for the inspection and preventive maintenance of the polar crane. The scope of this program includes the structural steel associated with the polar crane. The aging effect managed by the Inspection and Preventive Maintenance of the Polar Crane is a loss of material.

16.2.11 INSTRUMENT AIR QUALITY

With respect to license renewal, the Instrument Air Quality Program ensures that the instrument air supplied to components is maintained free of water and significant contaminants. The Instrument Air Quality Program applies to those components, within the scope of license renewal, supplied with instrument air where pressure boundary integrity is required for the component to perform its intended function. The aging effects requiring management addressed by the Instrument Air Quality Program are loss of material and cracking.

16.2.12 LEAKAGE DETECTION IN REACTOR BUILDING

Leakage detection in the reactor building monitors leakage to manage the consequences of cracking, loss of material, or loss of mechanical closure integrity. Leakage detection in the reactor building is focused on RCS leakage, but also includes other systems that have the potential to leak in the reactor building.

16.2.13 MAINTENANCE RULE

Maintenance Rule system and structural walk downs are conducted to detect and manage aging effects of structures and components within the scope of the license renewal. This includes coatings inspections of coated surfaces on structures and components. The Maintenance Rule is utilized to manage cracking, loss of material, loss of mechanical closure integrity, and change in material properties of structures and components within the scope of license renewal. This program applies only to external surfaces of the structures and components within the scope of the license renewal.

16.2.14 OIL ANALYSIS

The Oil Analysis Program ensures the oil environment in the mechanical systems is maintained to the quality required. Oil analysis program controls are credited as a program for maintaining oil systems free of contaminants (primarily water and particulates) thereby preserving an environment that is not conducive to corrosion. The scope of the Oil Analysis Program, with respect to license renewal, is limited to sampling activities and analysis on the auxiliary building electrical room chillers, emergency diesel generators, decay heat pumps, reactor building spray pumps, primary makeup pumps, diesel driven fire pump and engine, EFW pumps and turbine, the AAC diesel generator, and the control room chiller compressor. The Oil Analysis Program has been credited for ensuring the oil is free of water or contaminants. This manages the aging effects of cracking and loss of material.

16.2.15 PREVENTIVE MAINTENANCE

The purpose of the Preventive Maintenance Program is to perform preplanned, repetitive maintenance tasks on plant components and systems to extend equipment operating-life and to minimize the possibility of in-service component failures. The scope of the Preventive Maintenance Program, with regard to license renewal, is the preventive maintenance tasks credited with managing the aging effects listed below:

Preventive Maintenance Activity	Aging Effect
BWST internal inspection	Loss of material
BWST external inspection	Loss of material and loss of mechanical closure integrity
Reactor building ventilation cooling coil cleaning and inspection	Fouling, cracking* and loss of material
Hydrogen sampling system cabinet/heat exchanger cleaning, inspection, and lubrication	Fouling
Emergency fire diesel cooling water quarterly sampling for corrosion inhibitor	Loss of material
Emergency fire diesel intake air, exhaust air, lube oil and cooling water.	Loss of mechanical closure integrity
Emergency fire diesel inspection.	Loss of material, cracking, loss of mechanical closure integrity and fouling
Penetration room floor drain check valves inspection	Loss of material and cracking
Decay heat room drain valves inspection	Loss of material, cracking, and loss of mechanical closure integrity
EDG fuel oil tank inspection	Loss of material
EDG HVAC components inspection	Loss of material and loss of mechanical closure integrity
Control room ventilation inspections	Fouling and loss of material

Battery Charger and Penetration Room Cooler Cleaning and Inspection	Loss of material, cracking* and loss of mechanical closure integrity
Auxiliary Building Switchgear Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, cracking* and fouling
Auxiliary Building Decay Heat Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling
HPI Room Coolers Cleaning and Inspection	Loss of material, loss of mechanical closure integrity, and fouling

* cracking applies to inspection of expansion joint only

16.2.16 REACTOR BUILDING LEAK RATE TESTING

The Reactor Building Leak Rate Testing Program provides assurance that leakage from the reactor building does not exceed required maximum values for reactor building leakage. The Reactor Building Leak Rate Testing Program is comprised of Integrated Leak Rate Testing and Local Leak Rate Testing. The integrated leak rate test measures the primary reactor building overall integrated leakage rate. The scope of the integrated leak rate test is the entire reactor building. Integrated leak rate testing identifies loss of material or cracking. The local leak rate test measures the leakage across individual penetration components and determines the leakage of each penetration. Local leak rate testing identifies loss of material and cracking.

16.2.17 REACTOR BUILDING SUMP CLOSEOUT INSPECTION

The Reactor Building Sump Closeout Inspection detects significant degradation of the sump components and removes foreign objects that could impede suction from the sump. The scope of the Reactor Building Sump Closeout Inspection applies to reactor building sump, the area immediately surrounding the sump, the screening materials, and the equipment inside the sump. The aging effects addressed by the Reactor Building Sump Closeout Inspection are loss of material for the carbon steel components and cracking for stainless steel components.

16.2.18 REACTOR VESSEL INTEGRITY

The ANO-1 Reactor Vessel Integrity Program consists of the following five interrelated subprograms:

- Master Integrated Reactor Vessel Surveillance Program (MIRVP)

- Cavity Dosimetry Program
- Fluence and Uncertainty Calculations
- Pressure/Temperature Limits
- Monitoring Effective Full Power Years

The purpose of the MIRVP is to monitor reactor pressure vessel materials containing Linde 80 high copper beltline welds to determine the reduction of material toughness by neutron irradiation embrittlement. The purpose of the Cavity Dosimetry Program is to verify the accuracy of fluence calculations and to determine fluence uncertainty values. The purpose of the reactor vessel fluence and uncertainty calculations is to provide an accurate prediction of the actual reactor vessel accumulated neutron fast fluence value, for use in development of the pressure/temperature limit curves and pressurized thermal shock calculations. The purpose of the pressure/temperature limit curves is to establish the normal operating limits for the RCS. The pressure/temperature limit curves apply to the reactor vessel. The purpose of determining the EFPY is to accurately monitor and tabulate the accumulated operating time experienced by the reactor vessel. The reduction of material toughness by neutron irradiation embrittlement is the aging effect addressed by these five subprograms.

16.2.19 SERVICE WATER INTEGRITY

The service water integrity program ensures the service water system components continue to operate and perform their safety-related functions for the remaining life of ANO-1. The scope of the Service Water Integrity Program, with respect to license renewal, is limited to activities on service water system components and structures, including the emergency cooling pond. The Service Water Integrity Program has been credited with managing the following aging effects:

- The flow rate testing ensures the effects of fouling do not reduce flow rates below required values.
- The heat exchanger testing manages the aging effect of fouling by ensuring the heat exchangers can remove the necessary heat load.
- The thickness mapping and visual inspections manage the aging effects of loss of material from the service water components.
- Visual inspections of a sample of safety-related valves and heat exchangers manage the effect of cracking of the components.
- The service water bay inspection is credited for managing loss of material of the mechanical components in the service water bay.

- The ECP return line epoxy coating inspection manages loss of material in the ECP return line.

16.2.20 STEAM GENERATOR INTEGRITY

The Steam Generator Integrity Program ensures the steam generator integrity is maintained under normal operating, transient, and postulated accident conditions. The Steam Generator Integrity Program applies to the steam generator internals, tubing, and associated repair techniques and components, such as plugs and sleeves. The aging effects addressed by the Steam Generator Integrity Program are loss of material, cracking, and fouling.

16.2.21 SYSTEM AND COMPONENT MONITORING, INSPECTIONS, AND TESTING

16.2.21.1 Annual Emergency Cooling Pond Sounding

This program verifies the availability of a sufficient supply of cooling water in the emergency cooling pond to handle design basis accidents, with a concurrent loss of Lake Dardanelle. The scope includes the emergency cooling pond and surrounding structural components. The aging effect managed by this program is a loss of form of the emergency cooling pond due to sedimentation.

16.2.21.2 Battery Quarterly Surveillance

The battery rack inspections ensure their structural integrity. Seismically-qualified battery racks are within the scope. Battery racks and associated threaded fasteners are inspected for physical damage or abnormal deterioration including a loss of material.

16.2.21.3 Control Room Ventilation Testing

With respect to license renewal, the Control Room Ventilation Testing is credited as one of the programs to manage the aging effects. The control room ventilation testing applies to the control room emergency cooling coils. Fouling on the external surfaces of the cooling coil tubes is the aging effect managed by this program.

16.2.21.4 Core Flood Tank Monitoring

With respect to license renewal, the core flood tank monitoring provides a method to manage the aging effect of loss of material due to boric acid corrosion. The core flood tank monitoring applies to both core flood tanks. The loss of material due to boric acid corrosion on parts wetted by leaks from the core flood tanks may be detected through core flood tank monitoring.

16.2.21.5 Emergency Diesel Generator Testing and Inspections

With respect to license renewal, EDG Testing and Inspections provide a means of detecting aging effects associated with the various emergency diesel generator subsystems. The scope for these activities includes the emergency diesel generator assembly and associated support components. Loss of material is an aging effect for the carbon steel components in the EDG starting air system. Loss of material is identified as an aging effect for the unpainted carbon steel internal surfaces and the outer portion of the intake that could be wetted by rain. Loss of material and fouling are considered aging effects for the EDG intake air after coolers. Loss of material from the piping and muffler internal surfaces and from external surfaces exposed to the weather is an aging effect for the EDG exhaust components. Loss of material and fouling are aging effects for the lube oil coolers. The cooling water carbon steel components are susceptible to a limited loss of material from corrosion and the stainless steel components have the aging effect of cracking. Loss of material and fouling are aging effects for the cooling water heat exchangers. Since the portions of the subsystems on the engine are exposed to high vibration, loss of bolted closure integrity was identified as an aging effect for the skid mounted and connected components. The EDG day tanks are also drained and inspected internally for loss of material.

16.2.21.6 Emergency Feedwater Pump Testing

With respect to license renewal, the Emergency Feedwater Pump Testing is credited as one of the programs for managing the effects of aging. The scope includes the turbine and electric motor driven EFW pumps and associated components. Fouling in the system heat exchangers is the primary aging effect that this testing will identify. This testing also is credited with identifying the aging effects of loss of material and loss of mechanical closure integrity for system components.

16.2.21.7 NaOH Tank Level Monitoring

The NaOH tank level monitoring provides a method of detecting changes in the tank level that might indicate leakage from the NaOH tank or system. The NaOH tank level monitoring applies to the NaOH system components. This inspection is credited with managing the aging effects of loss of material, loss of mechanical closure integrity, and cracking.

16.2.21.8 Spent Fuel Pool Level Monitoring

The spent fuel pool level monitoring provides a method of detecting changes in the spent fuel pool level that might indicate cracks in the spent fuel pool liner. The spent fuel pool level monitoring applies to the detection of leakage through the spent fuel pool liner. Cracking of the spent fuel pool liner is the aging effect addressed by spent fuel pool level monitoring.

16.3 TIME-LIMITED AGING ANALYSIS ACTIVITIES

16.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

The reactor vessel is described in Sections 4.3.3. Time-limited aging analyses applicable to the reactor vessel are:

- neutron embrittlement of the beltline region, including pressurized thermal shock and Charpy upper-shelf energy reduction; and
- intergranular separation in the heat affected zone of low alloy steel under austenitic stainless steel weld cladding is addressed in Section 16.3.7.

The Reactor Vessel Integrity Program as described in Section 16.2.18 is being utilized to ensure that the time dependent parameters used in the time-limited aging analysis evaluations for pressurized thermal shock and Charpy upper-shelf energy reduction are tracked such that the time-limited aging analysis remains valid through the period of extended operation.

16.3.2 METAL FATIGUE

Cyclic loads are described in Section 4.1.2.4. For the extension of plant service-life from 40 years to 60 years, metal fatigue resulting from thermal transient cyclic loads is considered a time-limited aging analysis as defined by 10CFR54.21(c). The time-limited aging analysis requires metal fatigue evaluations to remain valid for the period of extended operation. This is achieved by maintaining adequate documentation of fatigue stress analyses to show the allowable design cycles of the RCS components for the applicable transient events and monitoring or tracking the actual operating cycles to ensure the allowable cycles are not exceeded. Fatigue evaluations are performed based on the design-allowable cycles specified in Table 4-8 and the NSSS vendor functional specification. As such, the fatigue evaluations and the fatigue life of the RCS components are dependent on the actual operating cycles and independent of the service-life of the plant. As long as the number of applicable transient cycles are below the design-allowable cycles, the fatigue evaluations originally performed for a 40-year plant service-life are applicable for the 60-year plant service-life. Transient cycle logging will be maintained to ensure the fatigue analysis assumptions remain valid during the period of extended operation.

An assessment of the potential for fatigue cracking of the pressurizer surge line, the makeup/high pressure injection (HPI) nozzles, and the decay heat removal ASME Class 1 piping has been conducted using the method and environmental fatigue data provided in NUREG/CR-5704 and NUREG/CR-6260.

This assessment has concluded that, prior to entering the period of extended operation, for each surge line and HPI nozzle and safe-end location that may exceed a CUF of 1.0 when considering environmental effects, an approach will be developed to show that the effects

of fatigue can be managed. The approach for addressing fatigue will include one or more of the following:

1. Further refinement of the fatigue analysis to lower the CUFs to below 1.0, or

2. Repair of the affected locations, or

3. Replacement of the affected locations, or

4. Management 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 Entergy Operations select Option 4 (i.e., inspection) to manage environmentally-assisted fatigue during the period of extended operation, inspection details such as scope, qualification, method, and frequency will be provided to the NRC prior to entering the period of extended operation.

16.3.3 ENVIRONMENTAL QUALIFICATION

Environmental qualification evaluations of electrical equipment are identified as time-limited aging analyses. Equipment covered by the EQ Program has been evaluated to determine if existing EQ aging analyses can be projected to the end of the period of extended operation by re-analysis or additional analysis. Qualification into the license renewal period will be treated the same as equipment initially qualified for 40 years or less. When aging analysis cannot justify a qualified life into the license renewal period, then the component or parts will be replaced prior to exceeding the qualified life in accordance with the EQ Program.

16.3.4 CONCRETE REACTOR BUILDING TENDON PRESTRESS

Loss of tendon prestress over the original license term is a time-limited aging analysis that was performed at the time of initial licensing and requires review for license renewal. Tendon lift off forces obtained during tendon surveillance are plotted and trended to assess the performance of the post-tensioning system. Acceptance criteria requires that actual forces during tendon surveillance be compared to the minimum required prestress forces of 1252 kips, 1274 kips, and 1233 kips for typical dome, vertical and hoop tendons, respectively. If trending indicates that minimum required prestress will not be met before the next tendon surveillance, corrective actions including re-tensioning, tendon replacement, or reanaylsis will be implemented. From the license renewal review, it was determined that the loss of prestress analysis is valid for the period of extended operation and will continue to be managed by tendon surveillance activities conducted under the ANO-1 ASME Section XI Inservice Inspection Program, IWL Inspections.

16.3.5 REACTOR BUILDING LINER PLATE FATIGUE ANALYSIS

Several thermal cycling conditions, which include annual outdoor temperature variations, changes in interior temperature during start-up and shutdown of the reactor coolant system, and DBA conditions were considered in the fatigue analysis of the liner plate. This analysis is a time-limited aging analysis. The projected number of cycles for these loadings for 60 years is bounded by the existing fatigue analysis. Therefore, the original design assumptions for addressing thermal fatigue of the liner plate and piping penetrations remain valid for the period of extended operation.

16.3.6 AGING OF BORAFLEX IN SPENT FUEL POOL RACKS

The Boraflex in the spent fuel pool racks is discussed in Section 9.6.2.3. Potential stressors for the Boraflex in the spent fuel pool racks include gamma flux, which changes the material characteristics of the base polymer, and chemical environment, from the exposure to borated water. Continued monitoring and analyses of the Boraflex degradation was committed to in the ANO response to Generic Letter 96-04 (0CAN109605). In order to ensure that the 5 percent subcriticality margin can be maintained for the life of the spent fuel storage racks, the existing coupon monitoring program will be continued. Spent fuel pool silica levels will continue to be monitored and silica evaluations will continue to be performed in order to confirm that the 5 percent subcriticality margin will be maintained through the next evaluation period. These reanalysis and sampling actions provide reasonable assurance that the effects of aging on the Boraflex in the spent fuel pool racks will be adequately managed for the period of extended operation. Additional information related to Boraflex is documented in NRC correspondence 1CAN090002.

16.3.7 REACTOR VESSEL UNDERCLAD CRACKING

Intergranular separations (underclad cracking) in low alloy steel heat-affected zones under austenitic stainless steel weld claddings were detected in SA-508, Class-2 reactor vessel forgings manufactured to a coarse grain practice and clad by high-heat-input submerged arc processes. BAW-10013 contains a fracture mechanics analysis that demonstrates the critical crack size required to initiate fast fracture is several orders of magnitude greater than the assumed maximum flaw size, plus predicted flaw growth due to design fatigue cycles. The flaw growth analysis was performed for a 40-year cyclic loading, and an end-of-life assessment of radiation embrittlement (i.e., fluence at 32 EFPY) was used to determine fracture toughness properties. The report concluded that the intergranular separation found in B&W vessels would not lead to vessel failure. This conclusion was accepted by the AEC. To cover the period of extended operation, an analysis was performed using current ASME Code requirements. This analysis is fully described in BAW-2251A. The maximum applied stress intensity factor for the emergency and faulted conditions occurs in the closure head to head flange region and the fracture toughness margin was determined to be 2.24, which is greater than the IWB-3612 acceptance

criterion of 1.41. It is therefore concluded that the postulated intergranular separations in the ANO-1 reactor vessel 508 Class-2 forgings are acceptable for continued safe operation through the period of extended operation.

16.3.8 REACTOR VESSEL NOZZLES – FLOW-INDUCED VIBRATION ENDURANCE LIMIT

Report BAW-10051, "Flow Induced Vibration Endurance Limit Assumptions," calculated stress values for the reactor vessel incore nozzles and compared them to endurance limit (stress) values. These endurance limit values were based on an assumed value of 10^{12} cycles for 40 years of operation. The number of fatigue cycles was extended for 60 years, and the component item stress values were compared to the recalculated endurance limit values and shown to be acceptable.

16.3.9 LEAK-BEFORE-BREAK

The successful application of leak-before-break to the main RCS piping is described in report BAW-1847 "The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSSS." This report provides the technical basis for evaluating postulated flaw growth in the main RCS piping under normal plus faulted loading conditions. LBB LOCA loadings are considered for the faulted analyses of fuel assemblies as described in Section 3.3.3.3.2.1, and LBB is credited in the Section 4.2.6.6 to justify pipe whip restraints being no longer required on the main RCS piping and may be removed as needed to facilitate maintenance or other activities. In addition, the analysis of reactor building internal pressure differentials following a LOCA applies LBB in the selection of breaks to be analyzed (see Section 14.2.2.5.5.2). The evaluation performed in FTI document 51-5000709-00, Assessment of TLAA Issues in LBB Analysis of RCS Primary Piping, demonstrates that the LBB analysis remains valid for the period of extended operation.

16.3.10 RCP MOTOR FLYWHEEL

Flaw growth analysis associated with the RCP motor flywheel is a time-limited aging analysis. The analysis for fatigue crack growth addresses the growth of pre-existing cracks. The crack growth analysis was performed for 4,000 startup/shutdown cycles for the RCP motors, which exceeds the number of design cycles by a factor of eight. Therefore, the existing crack growth analysis remains valid for the period of extended operation.