4 TIME-LIMITED AGING ANALYSES

4.1 Identification of Time-Limited Aging Analyses

4.1.1 Introduction

The applicant described its identification of time-limited aging analyses (TLAA) in Section 4.1.1, "Identification and Evaluation of Time-Limited Aging Analyses," of the LRA. The staff reviewed this section of the LRA to determine whether the applicant has identified the TLAAs as required by 10 CFR 54.21(c).

4.1.2 Summary of Technical Information in the Application

The applicant evaluated calculations for Plant Hatch against the six criteria specified in 10 CFR 54.3 to identify the TLAAs. As a result of this evaluation, the applicant identified the following TLAAs:

- piping stress analyses that consider thermal fatigue cycles defined by the life of the plant
- fatigue/stress analyses for the torus structure and nozzle connections
- piping wall thickness calculations that develop acceptable as-measured criteria for pipe walls on the basis of an anticipated corrosion rate that, in turn, is founded upon the life of the plant
- calculation of the corrosion allowance assumed for the reactor vessel
- environmental equipment qualification calculations that qualify electrical components for 40 years
- a containment penetration structural analysis that assumes a number of pressurization cycles over the 40-year life of the plant
- calculation of the reference temperature for nil-ductility for critical core region vessel materials accounting for radiation embrittlement (as required by 10 CFR Part 50, Appendix G)
- calculation of the end-of-life equivalent Charpy Upper-Shelf Energy margin (as required by 10 CFR Part 50, Appendix G) associated with the extended operating term
- analyses performed to demonstrate the acceptability of a technical alternative to the Code requirement for inspection of reactor pressure vessel circumferential welds
- change in the anticipated operating cycles of the MSIVs from the number of cycles assumed for 40 years in the Plant Hatch UFSAR

Pursuant to 10 CFR 54.21(c)(2), the applicant stated that it had not identified any exemptions granted under 10 CFR 50.12 that were based on a TLAA. The applicant did identify that a technical alternative (as defined in 10 CFR 50.55a(a)(3)(i)) to requirements to inspect circumferential welds

on the reactor pressure vessel had been approved. This TLAA is discussed in Section 4.6 of this SER.

4.1.3 Staff Evaluation

As indicated by the applicant, TLAAs are defined in 10 CFR 54.3 as analyses that meet the following six criteria:

- Involve systems, structures, and components within the scope of license renewal, as delineated in Section 54.4(a).
- Consider the effects of aging.
- Involve time-limited assumptions defined by the current operating term, for example, 40 years.
- Make a safety determination by determining which TLAAs are relevant.
- Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in Section 54.4(b).
- Ensure that the relevant TLAAs are contained or incorporated by reference in the CLB.

Table 4.1.1-1 of the LRA did not identify flaw growth analysis as a TLAA. Flaws in Class 1 components that exceed the size of allowable flaws defined in IWB-3500 of the ASME Code need not be repaired if they are analytically evaluated to the criteria in IWB-3600 of the ASME Code. The analytic evaluation requires that the applicant project the amount of flaw growth attributable to fatigue and stress corrosion cracking mechanisms, or both, where applicable, during a specified evaluation period. In RAI 4.1-1, the staff asked the applicant to identify all Class 1 components that have flaws that exceed the allowable flaw limits defined in IWB-3500 and that have been analytically evaluated to IWB-3600 of the ASME Code, and provide the results of the analyses that indicate whether the flaws will satisfy the criteria in IWB-3600 throughout the period of extended operation. In response, the applicant stated that the review of flaw growth analyses for Plant Hatch did not identify any that meet the definition of a TLAA per the criteria of 10 CFR 54.3. The applicant further indicated that most flaw evaluations were performed for a 40-month period, and no flaw evaluations were performed for a 40-year period. The staff agrees that evaluations for 40-month time periods do not constitute TLAAs per the definition in 10 CFR 54.3.

Table 4.1.1-1 of the LRA identifies piping stress analyses that consider thermal fatigue cycles as a TLAA. The table does not identify the fatigue analyses of other reactor coolant pressure boundary components or the reactor vessel internals as TLAAs. Section 4.2 of the LRA does address the reactor pressure vessel. In RAI 4.1-2, the staff asked the applicant to identify other components of the reactor coolant pressure boundary that have fatigue analyses. The staff also asked the applicant to describe the TLAAs that were performed to address fatigue for the reactor coolant pressure boundary components, except for the reactor vessel, that were not included in Table 4.1.1-1, and to describe the TLAA performed for the reactor vessel internals. The staff also requested that the applicant indicate how these TLAAs meet the requirements of 10 CFR 54.21(c). In response, the applicant stated that the criteria of BWRVIP-74 were used to determine which fatigue

analyses were sufficiently significant to constitute a TLAA. As indicated in the RAI, the applicant discussed the fatigue analysis of the reactor vessel internals in the UFSAR. In the SER issued in February, 2001, the staff requested that the applicant explain how the fatigue analysis of the vessel internals was found to be acceptable for the 60-year period. The staff also requested that the applicant identify any other components of the reactor coolant pressure boundary that had fatigue analyses, and explain how these analyses were found to be acceptable for the 60-year period. This was identified as part of Open Item 4.1.3-1 [4.1.3-1(a)].

The applicant provided a response to this open item by letter dated June 5, 2001. In the letter, the applicant indicated that the initial Plant Hatch vessel internals AMR noted that cracking due to fatigue was an aging effect requiring management and that the fatigue cumulative usage factor (CUF) calculation was a TLAA. The applicant's response also indicated that, subsequent to the development of the initial AMR, the end-of-life CUF was determined to be substantially less than 0.5. The applicant stated that since the end-of-life CUF was low, the fatigue calculation did not represent a TLAA. The staff disagrees with the applicant's premise that, because the calculated CUF was low, the fatigue calculation did not represent a TLAA. The applicant should have identified the vessel internals fatigue analysis as a TLAA in the LRA and described the disposition of the TLAA per the requirements of 10 CFR 54.21(c)(1). However, the applicant's current fatigue analysis of the vessel internals, which projects that the CUF will remain below 1.0 for the period of extended operation, provides an acceptable TLAA evaluation in accordance with the requirements of 10 CFR 54.21(c)(1)(ii). The applicant did not identify any other components of the reactor coolant pressure boundary that had fatigue analyses. Therefore, this part of Open Item 4.1.3-1 [4.1.3-1(a)] is closed.

Section 4.2.2 of the LRA contains a discussion of the Plant Hatch licensing-basis pipe break criteria. Part of the Plant Hatch pipe break criteria involves postulating pipe breaks at locations where the calculated fatigue usage exceeds a specified value. Although the applicant identified the fatigue cumulative usage factor (CUF) calculation as a TLAA, the applicant concluded that the pipe break criteria were only a screening mechanism and not a TLAA. The usage factor calculation used to identify postulated pipe break locations meets the definition of a TLAA, as specified in 10 CFR 54.3. In RAI 4.2-1, the staff asked the applicant to provide a description of a TLAA for the pipe break criteria at Plant Hatch, and describe how the TLAA meets the requirements of 10 CFR 54.21(c). In response, the applicant stated that it views the pipe break criteria to be selection criteria that establish a bounding set of locations for line break consideration. Although the staff agreed with the applicant's statement, the staff still considered pipe break postulations to be a TLAA because the fatigue calculation is a TLAA. Additionally, the NRC previously identified high-energy line break postulation founded on the fatigue CUF as a TLAA in accordance with 10 CFR 54.3 (60 FR 22480, May 8, 1995). Therefore, the staff requested that the applicant include pipe break postulations founded on the fatigue usage factor as a TLAA. This was identified as part of Open Item 4.1.3-1 [4.1.3-1(b)].

By letter dated September 5, 2001, the applicant responded to this open item. In the response to the open item, the applicant revised its LRA discussion of pipe break criteria to classify pipe break postulations based on fatigue CUF as TLAAs. The TLAA evaluation is discussed in Section 4.2.5 of the revised LRA. The licensing basis pipe break criteria required that breaks be postulated at piping locations where the calculated CUF exceeded 0.1. The applicant identified additional piping locations where the CUF criterion may be exceeded during the period of extended operation. The applicant proposed to monitor three bounding locations during the period of extended operation using its Component Cyclic or Transient Limit Program to address the TLAA. The applicant's proposed program, which involves monitoring a sample of bounding locations during the period of

extended operation, is an acceptable method to address the pipe break postulation TLAA in accordance with the requirements of 54.21(c)(1). If the CCTLP identifies a location where the usage criterion may be exceeded, then the applicant must take corrective action in accordance with the corrective action program. As part of the corrective action, other potential locations must be addressed: This part of Open Item 4.1.3-1 [4.1.3-1(b)] is closed.

4.1.4 Conclusions

The staff has reviewed the information in Section 4.1.1, "Identification and Evaluation of Time-Limited Aging Analyses," of the LRA. On the basis of that review, the staff concludes that the applicant has adequately identified the TLAAs as required by 10 CFR 54.21(c), and that no 10 CFR 50.12 exemptions have been granted on the basis of a TLAA, as defined in 10 CFR 54.3.

4.2 Pipe Stress

4.2.1 Introduction

The applicant described its evaluation of pipe stress TLAAs in Section 4.2, "Pipe Stress Time-Limited Aging Analyses" of the LRA. The staff reviewed this section of the LRA to determine whether the applicant has adequately evaluated the TLAAs as required by 10 CFR 54.21(c).

A metal component subjected to cyclic loads may fail at a load magnitude less than its ultimate load capacity as a result of metal fatigue, which initiates and propagates cracks in the material. The fatigue life of a component is a function of its material, its environment, and the number and magnitude of the applied cyclic loads. Fatigue was a design consideration for piping and components and, consequently, fatigue is part of the CLB for Plant Hatch. The applicant identified fatigue as TLAAs for piping stress analyses that consider thermal cycles defined by the life of the plant and fatigue/stress analyses for the torus structure and nozzle connections. The staff reviewed Section 4.2 of the LRA, which discusses thermal fatigue of piping and fatigue of the torus structure.

4.2.2 Summary of Technical Information in the Application

The applicant discusses the design criteria for thermal fatigue in Section 4.2.1 of the LRA. Class 1 piping was explicitly evaluated for thermal transients specified in the UFSAR. As indicated in Table 4.2.2-1 of the LRA, the Class 1 (RCS) piping at Unit 1 was designed to the United States of America Standard (USAS) B31.7 Class 1 criteria, and Unit 2 was designed to the criteria of ASME Code Section III, Subsection NB. The criteria of both codes require that the calculated fatigue CUF resulting from the thermal transients not exceed the specified code limit of 1.0. As indicated in Table 4.2.3-1 of the LRA, Non-Class 1 piping was designed to the criteria of either USAS B31.1, USAS B31.7 Class 2 and 3, or ASME Subsection NC and ND. The criteria of these codes specify a stress reduction factor to be applied to the allowable thermal bending stress range if the number of cycles exceeds 7,000.

The applicant discusses the evaluation of Class 1 components in Section 4.2.2 of the LRA. The applicant indicated that Class 1 fatigue TLAAs would be addressed by an aging management program, which is described in Section A.1.12 of the LRA. This aging management program monitors the CUF of specific bounding locations at Plant Hatch. Specifically, these locations include four components of the RPV; closure studs, the shell, the recirculation inlet nozzles, and the feedwater nozzles. In addition, the following Class 1 piping locations are monitored:

- Unit 1 RPV equalizer piping
- Unit 1 core spray piping (for replaced piping outside of the RPV)
- Unit 1 standby liquid control piping
- Unit 1 feedwater, HPCI, RCIC, and RWCU system piping
- Unit 1 main steam piping (loop B)
- Unit 2 main steam piping (loop D)
- Unit 2 feedwater piping
- Unit 2 steam condensate drainage piping

The applicant monitors these locations using its CCTL, which is discussed in Sections A.1.12 and B.1.12 of the LRA. The staff's evaluation of this program in contained in Section 3.1.12 of this SER.

The applicant also discusses the design criteria to postulate pipe break scenarios and Generic Safety Issue (GSI)-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life." The applicant states that the pipe break criteria are not a TLAA. The applicant relies on generic industry studies to address the environmental fatigue concerns identified in GSI-190.

The applicant discusses the evaluation of Non-Class 1 piping in Section 4.2.3 of the LRA. For Non-Class 1 piping, a stress reduction factor would have been applied if the number of equivalent full-temperature cycles exceeded 7,000. The applicant indicated that its review of the UFSAR, operations manual, and operating history indicated that the estimated number of full-temperature cycles that the Non-Class 1 piping would experience over 60 years is substantially less than the number assumed in the analyses.

The applicant discusses the evaluation of the torus structure in Section 4.2.4 of the LRA. Specifically, the applicant indicated that several calculations related to the torus structure constituted fatigue TLAAs. The applicant also indicated that a new analysis of the torus was performed to address fatigue for the period of extended operation.

4.2.3 Staff Evaluation

Components of the RCS were designed to codes that contained explicit criteria for the fatigue analysis. Consequently, the applicant identified fatigue analyses of some RCS components as TLAAs. In Section 4.1 of this SER, the staff questioned whether the applicant has identified all of the TLAAs. The staff reviewed the applicant's evaluation of the identified RCS components for compliance with the provisions of 10 CFR 54.21(c)(1).

The applicant monitors limiting locations in the RPV and RCS piping for fatigue usage through the use of its CCTLP. The applicant indicated that actual operating history was used to project a 60-year CUF for each unit. The applicant further indicated that all monitored locations are projected to have a CUF less than 1.0 after 60 years of operation. Even though the applicant projects that

the CUF of the limiting locations will not exceed 1.0 during the period of extended operation, the applicant relies on the CCTLP to monitor the CUF and manage fatigue in accordance with the provisions of 10 CFR 54.21(c)(1)(iii). The staff's evaluation of the CCTLP is contained in Section 3.1.12 of this SER.

The applicant's CCTLP tracks transients and cycles of RCS components that have explicit design basis transient cycles to ensure that these components stay within their design basis. Generic Safety Issue (GSI)-166, "Adequacy of the Fatigue Life of Metal Components," raised concerns regarding the conservatism of the fatigue curves used in the design of these components. Although GSI-166 was resolved for the current 40-year design life of operating plants, the staff initiated GSI-190 to address license renewal. The resolution of GSI-166 for the 40-year design life relied, in part, on conservatism in the existing CLB analyses. This conservatism included the number and magnitude of the cyclic loads postulated in the initial component design. A detailed discussion of the GSI-166 evaluation is contained in SECY-95-245, "Completion of the Fatigue Action Plan."

The staff's assessment for GSI-166 provides a basis for the current 40-year plant design life. However, the staff's assessment took credit for the conservatism in the CLB fatigue analyses for the 40-year plant life. The staff further indicated that its assessment could not be extrapolated beyond the current facility design life (40 years). Therefore, the GSI-166 resolution only applies to the fatigue accumulation for a 40-year design life.

The applicant's CCTLP tracks fatigue cycles of RCS components, and compares the cycles to those used in the CLB evaluation. GSI-166 and GSI-190 identified a concern regarding the conservatism of the CLB fatigue design curves. In SECY 95-245, the staff recommended not to backfit new fatigue criteria to current operating nuclear power plants. The recommendation was founded, in part, on an assessment of the conservatism in existing fatigue analyses of components at operating plants for the 40-year design life. The staff did recommend that a sample of components with high fatigue usage factors be evaluated for any period of extended operation.

By letter dated February 9, 1998, the Electric Power Research Institute (EPRI) submitted two technical reports dealing with the fatigue issue. EPRI Reports TR-107515, "Evaluation of Thermal Fatigue Effects on Systems Requiring Aging Management Review for License Renewal for the Calvert Cliffs Nuclear Power Plant," and TR-105759, "An Environmental Factor Approach to Account for Reactor Water Effects in Light Water Reactor Pressure Vessel and Piping Evaluations" were part of an industry attempt to resolve GSI-190. As recommended in SECY 95-245, the EPRI analyzed components with high usage factors, using environmental fatigue data. The staff has open technical concerns regarding the EPRI reports. The staff's technical concerns were transmitted to the Nuclear Energy Institute (NEI) by letter dated November 2, 1998. The NEI responded to the staff's concerns in a letter dated April 8, 1999. The staff submitted its assessment of the response in a letter to the NEI, dated August 6, 1999. As indicated in the staff's letter, the NEI response did not resolve all of the staff's technical concerns regarding the EPRI reports.

The applicant indicated that EPRI license renewal fatigue studies have demonstrated that sufficient conservatism exists in the design transient definitions to compensate for potential reactor water environmental effects for Plant Hatch. As discussed above, the staff does not agree with the contention that the EPRI fatigue studies have demonstrated that sufficient conservatism exists in the design transient definitions to compensate for potential reactor water environmental effects. Although the letter dated August 6, 1999 identified the staff's concerns regarding the EPRI procedure and its application to PWRs, the technical concerns regarding the application of the

Argonne National Laboratory (ANL) statistical correlations and strain threshold values are also relevant to BWRs. In addition to the concerns referenced above, the staff has additional concerns regarding the applicability of the EPRI BWR studies to Plant Hatch. EPRI Report TR-107943, "Environmental Fatigue Evaluations of Representative BWR Components," addressed a BWR-6 plant, and EPRI Report TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a Boiling Water Reactor Plant," used plant transient data from a newer vintage BWR-4 plant. In RAI 4.2-2, the staff requested that the applicant provide additional information regarding the use of the EPRI license renewal fatigue studies to resolve the environmental fatigue issue at Plant Hatch.

In response to the RAI, the applicant discussed its assessment of the impact of the environmental correction factors for carbon and low-alloy steels contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and those for austenitic stainless steels contained in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design of Austenitic Stainless Steels," on the results of the EPRI studies. As a result of its assessment, the applicant concluded that the correlations have been adequately accounted for via the conservatism of the design-basis transients.

The applicant indicated that EPRI Report TR-110356 contained studies that are directly applicable to Plant Hatch because they involved a BWR-4 that is identical to the Plant Hatch design. The only components evaluated in TR-110356 are the feedwater nozzle and the control rod drive penetration locations. However, the applicant indicated that both Plant Hatch units employ hydrogen water chemistry, whereas the plant in the EPRI study did not consider hydrogen water chemistry, which affects the level of dissolved oxygen in the primary system. Dissolved oxygen is an important factor in the environmental fatigue effects. The applicant stated that this issue was adequately addressed by its evaluation of the feedwater nozzle contained in EPRI Report TR-105759. It is not clear to the staff how the issue of hydrogen water chemistry was addressed in EPRI Report TR-105759. The applicant's response did not resolved the staff's concerns regarding the environmental fatigue issue at Plant Hatch.

The staff requested that the applicant provide an assessment of the six locations identified in NUREG/CR-6260, "Application of NUREG/CR-5999, 'Interim Fatigue Curves to Selected Nuclear Power Plant Components'," dated March 1995, for an older vintage BWR (BWR-4) considering the applicable environmental fatigue correlations provided in NUREG/CR-6583 and NUREG/CR-5704 reports for Plant Hatch Units 1 and 2. The applicant indicated that these locations are monitored by the CCTLP, and that the environmental factors have been adequately accounted for by the conservatism in the design basis transient definitions. On the basis of the above discussion, the staff did not agree with the applicant that environmental fatigue concerns regarding the six locations identified in NUREG/CR-6260 have been adequately addressed at Plant Hatch. The staff, therefore, requested that the applicant assess these six locations, considering applicable environmental fatigue correlations provided in NUREG/CR-6583 and NUREG/CR-5704, as applicable. This was identified as Open Item 4.2.3-1.

By letter dated September 5, 2001, the applicant provided a revised response to Open Item 4.2.3-1. The applicant committed to evaluate the locations identified in NUREG/CR-6260 using the applicable environmental fatigue correlations provided in NUREG/CR-6583 and NUREG/CR-5704. These locations are:

- Reactor Vessel (Lower Head to Shell Transition)

- Feedwater Nozzle
- Recirculation System (RHR Return Line Tee)
- Core Spray System (Nozzle and Safe End)
- Residual Heat Removal Line (Tapered Transition)
- Feedwater Line (RCIC Tee)

The applicant indicated that usage factor multipliers would be developed at each location to account for the environmental effects. The applicant further indicated that these environmental multipliers would be incorporated in the Hatch CCTLP. The applicant's CCTLP will monitor the CUF, which includes the environmental multipliers, at the six locations for comparison with the allowable CUF. The applicant's proposal adequately addresses the staff concern regarding environmental effects on fatigue usage and, therefore, Open Item 4.2.3-1 is considered closed.

The applicant discusses the TLAA for non-Class 1 piping in Section 4.2.3 of the LRA. The design code for non-Class 1 piping and tubing controls fatigue by limiting the allowable range of bending stresses resulting from the restraint of free-end expansion. The code provides for a reduction of the allowable stress range if the number of cycles exceeds 7000 full-range stress cycles. The applicant indicated that it estimated that the number of thermal cycles that non-Class 1 piping and tubing would encounter in 60 years of operation is substantially less than the number assumed in the original design. The applicant indicated that the current design basis for some piping and tubing is 14,000 cycles. In RAI 4.2-3, the staff requested that the applicant identify the piping and tubing that were designed for 14,000 cycles, and provide the basis for this specified number of cycles. In response, the applicant indicated that 14,000 cycles was assumed in design guides for instrumentation tubing and supports on the basis of a designer's rule-of-thumb approach. The applicant further indicated that the assumption is very conservative in that it implies a thermal cycle every 1.5 days over a 60-year operational life. The staff agrees with the applicant's assessment that the number of assumed cycles is conservative. The staff finds that the applicant's assessment satisfies the provisions of 10 CFR 54.21(c)(1)(i) by demonstrating that the analysis remains valid throughout the period of extended operation.

The applicant discusses its evaluation of the torus structure in Section 4.2.4 of the LRA. According to the applicant, several calculations that addressed fatigue of the torus structure met the criteria for a TLAA. The applicant indicated that a new analysis was necessary to address fatigue in the torus for the period of extended operation. The applicant indicated that the critical event leading to fatigue damage of the torus is the lifting of one or more main steam system safety relief valves (SRVs). The applicant proposed to manage fatigue of the torus by monitoring the number of SRV lifts in its CCTLP. The staff's evaluation of the CCTLP is contained in Section 3.1.12 of this SER.

4.2.4 Conclusion

The staff has reviewed the information in Section 4.2, "Pipe Stress Time-Limited Aging Analyses" of the LRA. On the basis of its review, the staff concludes that the applicant has adequately evaluated the pipe stress TLAAs, as required by 10 CFR 54.21(c)(1).

4.3 Corrosion Allowance

4.3.1 Introduction

The applicant described its evaluation of the corrosion allowance TLAA in Section 4.3, "Corrosion Allowance," of the LRA. The staff reviewed this section of the LRA to determine whether the applicant has adequately evaluated the TLAA as required by 10 CFR 54.21(c).

An allowance for corrosion was made in determining the appropriate thickness for pressure retaining components in the design of Plant Hatch. Only those analyses containing an assumption of a corrosion allowance that also tied the allowance to a 40-year operating life meet 10 CFR 54.3 Criterion 3. In the review of the Plant Hatch analyses, two scopes of supply are important. Specifically, these are the equipment designed and supplied by Bechtel, and the equipment designed and supplied by General Electric (GE).

4.3.2 Summary of the Technical Information in the Application

Bechtel Power Corporation Scope of Supply

The assumption of a corrosion allowance appears in calculations that confirm the pressure rating of piping and components. The piping specifications for both Plant Hatch units specify corrosion allowances for types of piping on the basis of material and environment. In most of the calculations reviewed, the corrosion allowance assumed was not tied to a 40-year life of the component. Additionally, corrosion rates were not identified (with specific exceptions discussed below). Many of the calculations used standard values from Table A104.2 of ASME B31.1. Once a required minimum wall thickness was calculated, the design often chose the next thicker component size (e.g., the next higher pipe schedule). For these reasons, calculations covering components in the Bechtel scope of supply generally do not meet the definition of a TLAA.

There is a subset of analyses that are the exception to the above paragraph. In the course of evaluating the residual heat removal service water system piping and the plant service water system piping in accordance with the NRC's Generic Letters 89-13, "Service Water System Problems Affecting Safety-Related Equipment," and 90-05, "Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2, and 3 Piping," Bechtel performed calculations to develop evaluation levels for measurements on the piping. These levels were founded, in part, on the expected thickness of a pipe and upon the predicted wear of that pipe for the remaining service life. In these analyses, the corrosion allowance from the pipe specification was assumed to be the maximum allowed for the 40-year service life of the piping. The corrosion rate thus defined is used in the calculations to predict the expected pipe thickness, and to develop the minimum acceptable as-found thickness of the pipe.

These calculations were instrumental in developing the inspection program for the residual heat removal and primary service water piping, much of which is within the scope of license renewal. The formulae used in the calculations have been retained in the inspection program procedure used at Plant Hatch.

Therefore, the plant service water and RHR service water inspection program uses one of two corrosion rates to predict the minimum acceptable measured pipe wall thickness. The first rate is defined by dividing the specified corrosion allowance by 40 years. The second rate is an observed

corrosion rate based upon several measurements of the pipe wall. The greater of the two corrosion rates is used to predict the acceptable minimum wall thickness. The action levels of the procedure are also based, in part, on the corrosion rate determined by the corrosion allowance.

The impact of an extended operating period on the inspection program is minimal. A change to the specification-based corrosion rate would not be conservative and is not necessary. Decreasing the corrosion rate (by dividing the current allowance by 60 rather than 40 years) is not appropriate, because a rate thus calculated would not be conservative.

The plant service water and RHR service water piping inspection program establishes screening levels for the piping. Therefore, the calculations are conservative for the extended term, and do not require revision. The plant service water and RHR service water inspection program will continue to manage the effects of aging (corrosion) for the extended license term, as required by 10 CFR 54.21(c)(1)(i) and (iii).

General Electric Scope of Supply

In reviewing the documents within the design records database, the applicant found no GE calculation or analysis that explicitly defined the corrosion allowance as a function of 40 years.

An extended operating period has a minimal impact on the inspection program. A change to the specification-based corrosion rate would not be conservative, and is not necessary. Decreasing the corrosion rate (by dividing the current allowance by 60 rather than 40 years) is not appropriate, because a rate thus calculated would not be conservative. Therefore, the applicant contracted GE to make a further determination within its scope of supply. The GE review developed the following conclusions about the stainless steel components, general piping, and reactor vessel. For austenitic stainless steel components in the Plant Hatch reactor system, the corrosion allowance was not explicitly calculated using a 40-year assumption. The corrosion rate for stainless steel under BWR conditions is very low, and the corrosion allowance will be adequate through the end of the renewal term. With respect to the reactor vessel, GE reviewed its internal communications, reports, and open literature to determine the method for calculating the Plant Hatch Unit 1 and 2 corrosion allowances. The GE review determined that, in one analysis, a time-dependent corrosion rate was used, and that the corrosion allowance was founded on a 40-year assumption for the service life of the vessel. Since this corrosion allowance was determined to meet all six criteria, the corrosion allowance is a TLAA. GE has evaluated the analysis in question and has determined that corrosion allowance assumed is adequate for operation through the end of the renewed license term, as required by 10 CFR 54.21(c)(1)(ii).

4.3.3 Staff Evaluation

Bechtel Power Corporation Scope of Supply

The staff has reviewed the applicant's discussion of the Bechtel Power Corporation scope of supply. Bechtel calculated the corrosion allowances on the basis of the type of piping and the environment, which the staff agrees is appropriate. The applicant reviewed the calculations, and generally found that standard values from Table A104.2 of ASME B31.1 were used. After calculating a minimum wall thickness, the next higher pipe schedule was selected. The staff agrees that this is standard practice. The applicant determined that the calculations in the Bechtel scope of supply generally do not meet the definition of a TLAA, and the staff agrees.

For the plant service water piping and residual heat removal service water piping, the applicant conducted TLAAs on the basis of a 40-year lifetime. The applicant divided the corrosion allowance by 40 years to develop a corrosion rate. This corrosion rate is used to determine the minimum pipe wall thickness at any time from the present to the end-of-life. On the basis of this calculation, the applicant developed an inspection plan for the residual heat removal and plant service water piping. Actual pipe wall thickness is measured and compared to the calculated wall thickness. The actual corrosion rate is calculated from the measured wall thickness and the time of service. The higher corrosion rate of the calculated value and the measured rate is used to predict the wall thickness at end-of-life. Since the corrosion allowance is somewhat arbitrary, the calculated corrosion rate is also arbitrary, and is not a particularly accurate predictor of future wall thickness. However, supplementing the calculated rate with measured rates gives credibility to the program. Therefore, the staff finds that this program is acceptable.

General Electric Scope of Supply

For the GE scope of supply, the only TLAA was for the service life of the vessel. GE has determined that the corrosion allowance is adequate for the extended period of operation. Since this conclusion is consistent with industry operating experience, the staff finds that the TLAA for the vessel is acceptable.

4.3.4 Conclusion

The staff has reviewed the information in Section 4.3, "Corrosion Allowance" of the LRA. On the basis of that review, the staff concludes that the applicant has adequately evaluated the corrosion allowance TLAA as required by 10 CFR 54.21(c)(1).

4.4 Environmental Qualification of Electrical Equipment

The Plant Hatch 10 CFR 50.49 Environmental Qualification (EQ) Program has been identified as a TLAA for the purposes of license renewal. The TLAA aspect of EQ encompasses all long-lived equipment whether active or passive, and each equipment qualification file for a long-lived component documents a TLAA.

The applicant described its TLAA for Environmental Qualification of Electrical Equipment in Section 4.4, "Environmental Qualification of Electrical Equipment," of the LRA. The staff reviewed this section of the LRA to determine whether the applicant provided adequate information to meet the requirements set forth in 10 CFR 54.21(c)(1) regarding an evaluation of EQ. The staff also reviewed Section 4.4.1 of the LRA to consider the applicant's resolution of Generic Safety Issue (GSI) 168, "Environmental Qualification of Electrical Components."

4.4.1 Summary of Technical Information in the Application

The Plant Hatch EQ TLAA evaluation implements 10 CFR 54.21 (c)(1) to demonstrate that (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. Following is a summary description of the EQ TLAA.

Scope of EQ Equipment

Based on a review of the Plant Hatch EQ documentation, the applicant identified electrical equipment important to safety that has a qualified life of at least 40 years, during which the electrical equipment can perform its intended functions during a LOCA or a high-energy line break (HELB) in the harsh environments of the containment and reactor building. The scope of equipment in the Plant Hatch EQ program is as follows:

- Safety-related (in accordance with the definition in 10 CFR 50.49(b), consistent with the Plant Hatch CLB) electrical equipment in a postulated harsh environment that is required to mitigate the consequences of the accident causing the harsh environment or whose subsequent failure can degrade safety systems or mislead the plant operator.

- Non-safety-related electrical equipment in a postulated harsh environment whose failure could impede a safety function or mislead the operator. The impact on emergency operation procedures should be considered in the failure analysis.

- Certain post-accident monitoring equipment located in a postulated harsh environment and designated as requiring qualification in the Regulatory Guide 1.97 section of Plant Hatch's response to Supplement 1 of NUREG-0737, "Clarification of TMI Action Plan Requirements."

EQ Process

The EQ process is controlled by the EQ Master List and the EQ procedures. The EQ Master List provides the following equipment information:

- plant tag number of the equipment

- the manufacturer and model or series number of the equipment

- the building, floor elevation, and specific location of the equipment

- the Qualification Data Package (QDP) which addresses qualification and maintaining qualification of equipment

The EQ Installation/Maintenance Procedure Outline (I/MPO) specifically addresses the following:

- maintenance required to maintain equipment qualification

- qualified life of the equipment, any component part to be replaced, and the replacement interval (e.g., replace cover o-ring every 18 months)

- sealing of the equipment cable entrance to prevent moisture intrusion, as required

- installation and mounting configurations required to maintain qualification

- shelf life or storage requirements

- information on procuring and reordering equipment

Replacement Equipment

Prior to the expiration of the qualified life of a piece of EQ equipment, the Plant Hatch work management system generates a maintenance work order to alert plant personnel that the equipment is scheduled for replacement in the near future with the following available options:

- replace the existing component with an identical component
- replace the equipment with different equipment which is already evaluated under the EQ program
- replace the equipment with different equipment which is not currently evaluated under the EQ program (this option requires an equipment review, a function review, and an EQ review)
- reanalyze qualified life calculations to extend the qualified life if excess conservatism exists in the original qualified life calculation. Conservatism may exist in parameters such as the assumed ambient temperature of the equipment, an unrealistically low activation energy, or in the application of the equipment. The reanalysis is documented in the EQ central file. The guidelines in EPRI TR-104873, "Methodologies and Procedures to Optimize Environmental Qualification Replacement Intervals," are followed. Reanalysis is performed at Plant Hatch as follows:
    - Analytical Methods - The Arrhenius methodology is the thermal model used to reanalyze qualified life calculations. During normal operations, equipment is only subjected to ambient humidity levels (20-90 percent). Environmentally qualified equipment is typically sealed and cable insulation is protected from occasional inadvertent spray. Exposure to moisture from leaks is investigated on a case-by-case basis. The analytical method used for radiation reanalysis identified the 40-year radiation dose from the EQ criteria manual for the area where the equipment is installed, multiplied that value by the ratio of the evaluation period divided by 40 years (e.g., for license renewal 60 years/40 years, or 1.5), and added the applicable accident radiation dose to obtain the total integrated dose for the equipment. Plant Hatch has specifically assessed the impact of life extension from 40 to 60 years on the EQ radiation exposures for both units.
    - Data Collection and Reduction Methods - Reducing excess conservatism in the equipment service temperatures used in existing analyses is the chief purpose of reanalysis. Temperature data for a reanalysis is obtained from actual temperature measurements in the area around the equipment being reanalyzed. Temperature measurements can be obtained from monitors used for technical specification compliance, from other installed monitors, or from temperature sensors on specific components. The measurements can also be taken by plant operators during surveillance rounds. A representative number of temperature measurements is mathematically reduced to arrive at a temperature for the reanalysis. A reanalysis may use the actual calculated temperature, or may use the calculated temperature to show conservatism in the design temperature.

- Underlying Assumptions - Conservatism in the EQ equipment qualification analyses has been maintained sufficiently to absorb environmental changes due to plant modifications and events. Major plant modifications or events of sufficient duration (such as power uprates) to change temperature, pressure, and/or radiation values used in the underlying assumptions or in the EQ calculations are addressed in the design phase, prior to implementation of the plant modification, or operational change (the process by which changes to the underlying assumptions are made is discussed below under "Plant Environmental Changes.")

- Acceptance Criteria and Corrective Actions - Adequate margin as described in IEEE Std. 323-1974 and the Division of Operating Reactor Guidelines, is maintained in all reanalyses, or adequate justification reducing margin is provided. If the reanalysis does not maintain adequate margin and less margin cannot be justified, the equipment qualification is not extended and the equipment is replaced as scheduled prior to the expiration of the existing qualification.

Refurbishment of Environmentally Qualified Electrical Equipment

Equipment in need of refurbishment is typically replaced with new equipment or previously refurbished equipment taken out of storage. The removed equipment is then discarded or refurbished and placed in storage. Qualified equipment is required to be refurbished before it can be put back in storage. Refurbishment is performed in a manner that preserves the equipment's qualification. "Soft" items, such as gaskets, seals, and wires, which have a limited life, are typically replaced.

The manufacturer and model of replacement parts with an EQ-limited life are identified in the I/MPO, EQ maintenance procedures, and vendor manuals for environmentally qualified equipment. The documentation includes guidance on the shelf life of refurbished equipment.

Procurement of EQ Equipment

Procurement policies and criteria for environmentally qualified equipment are controlled by site procedures and the Nuclear Quality Assurance Program. Procurement of like-for-like replacement of environmentally qualified equipment is controlled so that the procured equipment is as good as, or better than, the original equipment. The procurement process also assures that applicable performance requirements and qualification criteria are met. The component's QDP contains procurement information such as the manufacturer or vendor, test reports to be referenced on the requisition, and equipment specifications.

Specifications for procurement are reviewed, and test plans are reviewed and approved prior to testing to assure compliance with the specifications. New test reports are evaluated and inserted into the QDP, and the EQ Master List is updated.

Plant Environmental Changes

Engineering Specification SS-2102-238, documents plant environmental conditions for both normal and accident conditions. The harsh environment areas of the plant for LOCAs, HELBs, and radiation are identified in accordance with the CLB. The Plant Hatch EQ central file contains temperature and pressure profiles for the various accident scenarios, including worst-case

composite accident profiles for the harsh environments of the containment and reactor building. The central file also contains the supporting calculations for these accident profiles and total integrated radiation doses. All specifications, calculations, and other central file documents are controlled documents.

Measurements of critical parameters, such as containment temperatures for technical specifications, are taken on an ongoing basis. Changes in environmental parameters are reviewed when found or anticipated as a result of an impending design change. When a significant environmental change is identified, a review of the qualification of affected environmentally qualified equipment is performed and applicable changes are made to the equipment's qualified life and QDP documentation. The EQ calculations, specifications, and accident profiles are revised, as appropriate, to reflect the new operating conditions.

<u>EQ Generic Safety Issue</u>

GSI-168 was developed to address environmental qualification of electrical equipment. The staff guidance to the industry (letter dated June 2, 1998 from NRC (Grimes) to NEI (Walters)) states:

- GSI-168 issues have not been identified to a point that a license renewal applicant can be reasonably expected to address these issues, specifically at this time; and

- An acceptable approach is to provide a technical rationale demonstrating that the CLB for EQ will be maintained in the period of extended operation.

For the purpose of license renewal, as discussed in the SOC (60 FR 22484, May 8, 1995), there are three options for addressing issues associated with a GSI:

- If the issue is resolved before the renewal application is submitted, the applicant can incorporate the resolution into the application.

- An applicant can submit a technical rationale that demonstrates that the CLB will be maintained through the period of extended operation until one or more reasonable options become available to adequately manage the effects of aging.

- An applicant can develop a plant-specific aging management program that incorporates a resolution to the aging issue.

To address issues associated with GSI-168, the applicant has chosen to pursue the second approach. The applicant will continue to manage the effects of aging in accordance with the CLB and considers the evaluation of the EQ TLAA in Section 4.4 of the LRA to be the technical rationale that demonstrates that the CLB will be maintained until some later point in the period of extended operation, when one or more reasonable options become available to adequately manage the effects of aging.

4.4.2 Staff Evaluation

In accordance with 10 CFR 54.21(c)(1), the staff reviewed Section 4.4 of the LRA to determine whether the applicant provided adequate information to meet the requirements that (i) the analyses remain valid for the period of extended operation; (ii) the analyses have been projected to the end

of the period of extended operation; or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. The staff also reviewed the treatment of GSI-168 in Section 4.4 of the LRA. After completing the initial review, the staff issued RAIs on July 28, 2000, and met with the applicant on August 23, 2000, to discuss RAIs 4.4-1 and 4.4-2 in the EQ TLAA area. The staff received the applicant's responses to the RAIs by letter dated October 10, 2000.

The applicant is using standard approved EQ methodologies and acceptance criteria, as defined by NRC Bulletin 79-01B, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines), including Supplements 1, 2, and 3; NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1; 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"; RG 1.89, " Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Revision 1; various NRC generic letters and information notices; and NRC safety evaluation reports on EQ. The current actions for short-lived environmentally qualified equipment are also acceptable for long-lived EQ equipment. As discussed below, the staff concurs with the applicant's EQ methodology.

The applicant is implementing 10 CFR 54.21(c)(1)(i), (ii), and (iii) for evaluating the EQ TLAA. The staff reviewed the following aspects of the applicant's EQ TLAA methodology:

- Scope of EQ program
- EQ process
    - Original qualification basis
    - EQ master list
    - EQ maintenance
    - Replacement of equipment
    - Replace the existing equipment with identical equipment
    - Replace the equipment with different equipment currently evaluated under the EQ program
    - Replace the equipment with different equipment not currently evaluated under the EQ program
    - Reanalyze the qualified life calculation
    - Refurbishment of environmentally qualified equipment
    - Procurement of environmentally qualified equipment
    - Plant environmental changes

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(i)

Section 4.4.5 of the LRA lists various commodity types based on Option (i) of 10 CFR 54.21(c)(1) whose analyses remain valid for the period of extended operation. In its response to RAI 4.4-1, the applicant provided thermal and radiation summaries for 38 commodity types that are based on Option (i). The staff reviewed the analyses and finds the demonstration of 10 CFR 54.21(c)(1)(i) for these commodity types to be acceptable for the period of extended operation.

TLAA Demonstration for Option 10 CFR 54.21(c)(1)(ii)

Section 4.4.5 of the LRA lists various commodity types based on Option (ii) of 10 CFR 54.21(c)(1) whose analyses have been projected to the end of the period of extended operation. During a meeting on August 23, 2000, the staff reviewed the EQ calculations for projecting the qualified lives of the following sample of commodity types to the end of the period of extended operation:

- Limitorque SB, SMB Actuators, AC Service
- General Electric F01 Electrical Penetration Assemblies
- Amphenol Type HN Plug Connectors
- States ZWM and NT Series Terminal Blocks
- Raychem Breakout/Scotchcast 9 Potting Compound
- AMP Special Ind. Insulated/Uninsulated Terminals and Splices
- Okonite Low/Medium Voltage Instrumentation, Control, and Power Cables
- Okonite T-95 Insulating and No. 35 Jacketing Tapes/Cement
- Anaconda Low Voltage Instrumentation, Control, and Power Cables
- GE RHR and Core Spray Pump Motors
- Brand-Rex Low Voltage Instrumentation, Control, and Power Cables and Internal Panel Wiring
- Conax Buffalo Electrical Penetration Assemblies
- Eaton (Samuel Moore) Instrumentation and Thermocouple Cables
- Reliance Motors FNA-6856 and 6857

Based on the staff's review of the applicant's thermal and radiation summaries and the EQ calculations that were reviewed during the August 23, 2000, meeting, the staff finds the demonstration of 10 CFR 54.21(c)(1)(ii) to be acceptable for the Option (ii) commodity types listed in Section 4.4.5 of the LRA.

TLAA Demonstration for the 10 CFR 54.21(c)(1)(iii) Option

Section 4.4.5 of the LRA lists various commodity types based on Option (iii) of 10 CFR 54.21(c)(1) on which the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. For Option (iii) commodity types whose qualified lives could not be extended significantly, the Plant Hatch EQ program and the associated site administrative controls have the necessary elements to ensure that the effects of aging on the intended function(s) of the qualified equipment will be adequately managed for the period of extended operation. For EQ components that cannot be qualified to the end of the period of extended operation, aging effects will continue to be managed in accordance with the current licensing basis, which requires that equipment be replaced or refurbished at the end of its qualified life unless ongoing qualification demonstrates that the item has additional life. The staff finds this approach to be an acceptable demonstration of 10 CFR 54.21(c)(1)(iii) for managing the effects of aging on environmentally qualified components for the period of extended operation.

GSI-168 Finding

The staff finds that the applicant's approach to resolving GSI-168 for license renewal (i.e., continuing to manage the effects of aging in accordance with the CLB until one or more reasonable options become available to adequately manage the effects of aging) is consistent with the June 2, 1998, staff guidance to industry.

4.4.3 Conclusion

The staff has reviewed the EQ TLAA information in Section 4.4 of the Plant Hatch LRA, the additional information provided in the August 23, 2000, meeting on EQ between the staff and the applicant, and the October 10, 2000, response to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has demonstrated, pursuant to 10 CFR 54.21(c)(1), that, for TLAAs related to environmental qualification of electrical equipment, (i) the analyses remain valid for the period of extended operation, (ii) the analyses have been projected to the end of the period of extended operation, or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation. In addition, the staff finds the applicant's approach to resolving GSI-168 acceptable.

4.5 Containment Penetration Pressurization Cycles

In Section 4.5 of the LRA, the applicant described the time-limited effect of pressurization cycles on the design of containment penetrations. The staff reviewed this section of the LRA to determine whether the applicant has demonstrated that the effects of aging on the containment penetrations will be adequately managed during the period of extended operation, pursuant to10 CFR 54.21(c)(1).

4.5.1 Summary of Technical Information in the Application

The applicant identified one containment penetration structural analysis for Plant Hatch that assumed a number of pressurization cycles over a 40-year period. This calculation was determined to meet the definition of a TLAA, as stated in 10 CFR 54.3 and Section 4.1 of this SER. The applicant also stated that the architect-engineer performed a structural analysis to determine the acceptability of certain types of pipe-to-penetration welds using backing rings. The effects of the

pressurization cycles on these calculations were stated as being minimal. The applicant also stated that the calculation had been extended to 60 years of operation without a change to plant equipment, on the basis of Criterion (ii) of 10 CFR 54.21(c)(1).

4.5.2 Staff Evaluation

The staff reviewed the information provided in Section 4.5 of the LRA regarding fatigue analyses of containment penetrations, and concluded that additional information was needed to complete its review. The staff issued RAIs by letter dated July 28, 2000. By letter dated October 10, 2000, the applicant provided responses to the RAIs. The staff has evaluated the applicant's responses, as described in the following paragraphs.

In RAI 4.5-1, the staff requested that the applicant identify the containment penetration for which the structural analysis assumed a number of pressurization cycles for 40 years. The RAI requested that the applicant provide the location of each penetration, the number of pressurization cycles that each was assumed to undergo during the current licensing term, the actual cycles that have been experienced, and the number of cycles that are expected until the end of the extended period of operation. Since containment penetrations also experience thermal cycling as a result of plant operation, the staff also requested that the applicant provide the number of thermal cycles for which each penetration had been evaluated. In addition, the staff requested that the applicant provide a summary of the structural analysis that was performed to demonstrate the acceptability of the pipe-to-penetration welds using backing rings.

In its response, the applicant states that the calculation applies to the Class B weld of the main steam penetration assembly to the containment, and justifies the use of a backing ring for that type and location of weld. In the original calculation, the applicant assumed 40 pressurization cycles to full design pressure, and that number was later revised to consider 60 pressurization cycles to full design pressure. The applicant stated that this assumption is conservative, and that it had therefore demonstrated the acceptability of the analysis in accordance with 10 CFR 54.21(c)(1)(ii). In addition, the response indicated that the calculation applies to a Class B weld that is referenced in ASME Section III, N-415.1, 1968 Edition, "Vessels Not Requiring Analysis for Cyclic Operation." Reference to N-415.1 indicates that the stresses attributable to the pressurization cycles were found to meet the limiting stress criterion, which does not require a fatigue analysis under the provisions of this section. By letter dated January 24, 2001, the applicant submitted additional (proprietary) information, which provided justification for concluding that thermal cycling of the penetration assembly does not represent a significant loading condition, which would require a fatigue analysis under the provisions of ASME Section III, N-415.1. The staff reviewed this information and concluded that the applicant has demonstrated that this TLAA for the containment penetrations will remain valid for the period of extended operation. The staff therefore finds the response to RAI 4.5-1 acceptable and considers this concern resolved.

In RAI 4.5-2, the staff requested that the applicant provide information regarding the effect of thermal cycling on the drywell and torus vent line penetrations and penetration bellows (including vent line bellows), and dissimilar metal welds resulting from reactor mode changes and other transients, pressurization pulses during SRV discharges, and pressure cycles during leak testing. In its response, the applicant stated that the information requested in this RAI pertaining to containment torus penetrations is summarized in the design analysis addressing fatigue in the torus for the license renewal period ("Hatch Units 1 and 2 Torus Fatigue Analysis Report, REA HT-98674 Response", Revision 0, Southern Company Services, Inc., Nuclear Engineering and Regulatory

Support, April 1999). In Section 4.2.4 of the LRA, the applicant stated that the CLB fatigue calculations for the torus structure were reviewed and, on this basis, the applicant determined that a new analysis was necessary to address fatigue in the torus for the extended license term. The analysis required an extensive and detailed review of pressure and thermal transients for the torus. By letter dated January 24, 2001, the applicant provided a (proprietary) summary of this analysis. The staff has reviewed this information and concludes that the applicant has demonstrated satisfactorily that the fatigue adequacy of the Unit 1 and Unit 2 torus penetrations under the CLB transient operating conditions will be maintained during the period of extended operation. The applicant also addressed the fatigue adequacy of the drywell penetrations by referencing EPRI report TR-103840 "BWR Containments License Renewal Industry Report; Revision 1" July 1994, which indicates that fatigue of these penetrations subject to the CLB transient operating conditions will be minimal for the period of extended operation. The staff finds the response to RAI 4.5-2 acceptable, and considers this concern resolved.

In RAI 4.5-3 the staff requested that the applicant provide a list of the containment penetrations with pipe-to-penetration welds. In RAI 4.5-4, the staff requested that the applicant provide justification for not performing fatigue TLAAs on containment penetrations with pipe-to-penetration welds that are susceptible to combined pressurization cycles and plant operational thermal expansion cycles. In its response, the applicant stated that the Unit 1 and 2 current licensing bases were reviewed, and that no specific analyses on this subject were found that met the criteria of 10 CFR 54.3 for a fatigue TLAA. However, the applicant indicated that fatigue of the ASME Code Class 1 welds is bounded by the locations monitored in the component cycle and transient limit program. The applicant further stated that the fatigue of the Non-Class 1 welds is bounded by the number of cycles assumed in the original analysis. The staff concurs with the applicant's response, and considers the concerns stated in RAIs 4.5-3 and 4.5-4 resolved.

4.5.3 Conclusion

The staff has reviewed the information in Section 4.5 "Containment Penetration Pressurization Cycles" of the LRA, the applicant's responses to the staff's RAIs, and the information provided to the staff by letter dated January 24, 2001. On the basis of this review, and pursuant to 10 CFR 54.21(c)(1), the staff concludes that the applicant has adequately evaluated the containment penetration pressurization cycles TLAA.

4.6 Time-Limited Aging Analyses for the Reactor Vessel

4.6.1 Summary of Technical Information in the Application

Neutron Irradiation Embrittlement

Neutron irradiation causes a decrease in the Charpy upper-shelf energy (USE) and an increase in the adjusted reference temperature (ART) of the RPV beltline materials. The ART impacts the plant's pressure-temperature (P-T) limits and RPV integrity evaluations. BWRVIP-74 has performed integrity evaluations of BWR RPV circumferentially oriented welds and BWR RPV axially oriented welds. Therefore, in order to demonstrate that neutron embrittlement does not significantly impact RPV integrity during the license renewal term, BWRs must evaluate the impact of neutron irradiation on the Charpy USE, ART, RPV circumferential welds, and RPV axial welds.

Charpy (USE)

By letter dated April 30, 1993, the Boiling Water Reactor Owner's Group (BWROG) submitted a topical report entitled "10 CFR Part 50 Appendix G Equivalent Margins Analysis for Low Upper Shelf Energy in BWR/2 Through BWR/6 Vessels," to document that BWR RPVs could meet the margins of safety against fracture equivalent to those required by Appendix G of the ASME Code for Charpy USE values less than 50 ft-lb. GE performed an update to the USE equivalent margins analysis, which is documented in EPRI TR-113596, "BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, dated September 1999. This updated analysis incorporates the effects of irradiation for 54 effective full-power years (EFPY), which corresponds to 60 years of operation at 90-percent power. The updated analysis determined that the generic materials considered will maintain the margins for USE required by Appendix G of 10 CFR Part 50. GE reviewed the updated generic analyses with respect to applicability for the Plant Hatch license renewal term. This review is documented in an evaluation performed by GE in GENE B11-00827-00-01, "Plant Hatch Units 1 and 2 Reactor Pressure Vessel Pressure/Temperature Limits License Renewal Evaluation," General Electric Company, dated March 1999. GE determined that the generic analyses are applicable and that, for 54 EFPY, the critical materials would retain sufficient USE to satisfy the requirements to 10 CFR Part 50 Appendix G.

Reference Temperature Adjustments

GE performed a plant-specific analysis of the ART for Plant Hatch in GENE B11-00827-00-01, using the criteria defined in EPRI TR-113596. The GE analysis for Plant Hatch considers the effect of neutron embrittlement for 54 EFPY. The analysis includes new sets of reactor operating pressure and temperature curves. The results of the analysis indicate that for both units, the ART will be less than 200 °F.

Circumferential RPV Weld Inspection Relief

The BWRVIP provided the technical bases supporting the elimination of RPV circumferential welds from the inservice inspection programs for BWRs in EPRI TR-113596. These technical bases are approved for the current license term, and are applicable to Plant Hatch.

Appendix E to the NRC's "Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report (TAC No. M93925)," USNRC, dated July 28, 1998, documents an evaluation of the impact of license renewal from 32 EFPY to 64 EFPY on the conditional probability of vessel failure. That SER reports that the frequency of cold overpressurization events results in a total vessel failure probability of approximately 5×10^{-7}. The SER conservatively evaluates an operating period of 10 EFPY greater than what is realistically expected for a 20-year license renewal term (i.e. 48 to 54 EFPY.) Therefore, this analysis provides a basis for BWRVIP-05 to be approved as a technical alternative to the current inservice inspection requirements of ASME Section XI for volumetric examination of the circumferential welds as they may apply in the license renewal period.

Axially Oriented RPV Welds

The staff's SER, contained in a letter dated March 7, 2000, to Carl Terry, BWRVIP Chairman, discusses the staff's concern related to the RPV failure frequency of axial welds, and the BWRVIP's analysis of the failure frequency. The SER indicates that the RPV failure frequency attributable to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is below 5 x

10^{-6} per reactor year, given the assumptions regarding flaw density, distribution, and location described in the SER. Since the BWRVIP analysis was generic, the applicant provided plant-specific information in response to RAI 4.6-2 to demonstrate that the Plant Hatch beltline materials meet the criteria specified in the report.

4.6.2 Staff Evaluation

Neutron Irradiation Embrittlement

Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary of light-water nuclear power reactors. It also provides adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. For the RPV, this appendix requires an evaluation of the Charpy USE and ART to determine pressure-temperature limits for the RPV. Neutron irradiation causes a decrease in the Charpy USE and an increase in the adjusted reference temperature of the RPV beltline materials. The staff's evaluation of the impact of irradiation on the Charpy USE, adjusted reference temperature, RPV circumferential weld, and RPV axial weld integrity analysis is discussed in this section. Since each of these evaluations are dependent upon the neutron fluence received by the RPV, neutron fluence is also discussed in this section.

Charpy (USE)

Section IV.A.1a. of Appendix G to 10 CFR Part 50 requires, in part, that RPV beltline materials must have Charpy USE in the transverse direction for base metal, and along the weld for weld material of no less than 50 ft-lb (68J), unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy USE will provide margins of safety against fracture equivalent to those required by Appendix G to Section XI of the ASME Code.

By letter dated April 30, 1993, the BWROG submitted a topical report entitled "10 CFR Part 50 Appendix G Equivalent Margins Analysis for Low Upper-Shelf Energy in BWR/2 Through BWR/6 Vessels," to document that BWR RPVs could meet the margins of safety against fracture equivalent to those required by Appendix G to the ASME Code for Charpy USE values less than 50 ft-lb. In a letter dated December 8, 1993, the staff concluded that the topical report demonstrates that the evaluated materials have the margins of safety against fracture equivalent to Appendix G to the ASME Code, in accordance with Appendix G to 10 CFR Part 50. In this report, the BWROG derived through statistical analysis the initial USE values for materials that originally did not have documented Charpy USE values. Using these statistically derived Charpy USE values, the BWROG predicted the end-of life (40 years of operation) USE values in accordance with Regulatory Guide (RG) 1.99, Revision 2. According to this RG, the decrease in USE is dependent upon the amount of copper in the material and the neutron fluence predicted for the material. The BWROG analysis determined that the minimum allowable Charpy USE in the transverse direction for base metal and along the weld for weld metal was 35 ft-lb.

GE performed an update to the USE equivalent margins analysis, which is documented in EPRI TR-113596, "BWR Vessel and Internals Project BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, dated September 1999. EPRI TR-113596 provides a bounding Charpy USE for BWR plants for 54 effective full-power years (EFPY). Specifically, the bounding analysis for Plant Hatch-type plants (BWR/4) indicates that at 54 EFPY, the Charpy USE in the

transverse direction for plates would be at least 45 ft-lb, and the Charpy USE for the non-Linde 80 submerged arc welds (SAWs) would be at least 43 ft-lb. Since these values are greater than the minimum allowable Charpy USE of 35 ft-lb, these materials would have margins of safety against fracture equivalent to Appendix G to the ASME Code. Since this was a generic analysis, the applicant should provide plant-specific information to demonstrate that the Plant Hatch beltline materials meet the criteria specified in the report.

The analysis in EPRI TR-113596 utilized an unirradiated Charpy USE in the longitudinal direction of 91 ft-lb for BWR/3-6 plates, and 70.5 ft-lb for non-Linde 80 submerged arc welds. The value for the plates is the lowest value from the database, and is less than the lower 95/95 confidence value. The value for the non-Linde 80 submerged arc welds is the value corresponding to the lower 95/95 confidence value. Since these values are statistically determined with at least 95/95 confidence, the values may be used in the evaluation of Charpy USE.

The analysis in EPRI TR-113596 determined the reduction in the unirradiated Charpy USE resulting from neutron radiation using the methodology in RG 1.99, Revision 2. Using this methodology with a correction factor of 65 percent for conversion of the longitudinal properties to transverse properties, the lowest irradiated Charpy USE at 54 EFPY for all BWR/3-6 plates is projected to be 45 ft-lb. The correction factor for specimen orientation in plates is predicated on NRC Branch Technical Position MTEB 5-2, "Fracture Toughness Requirements." July 1981. Using the RG methodology, the lowest irradiated Charpy USE at 54 EFPY for BWR non-Linde 80 submerged arc welds is projected to be 43 ft-lb. EPRI TR-113596 indicates that the percent reduction in Charpy USE for the limiting BWR/3-6 plates and BWR non-Linde 80 submerged arc welds is 23.5 percent and 39 percent, respectively. To demonstrate that the Plant Hatch beltline materials meet the criteria specified in the report, the applicant should demonstrate that the percent reduction in Charpy USE for its beltline materials is less than those specified for the limiting BWR/3-6 plates and the non-Linde 80 submerged arc welds, and that the percent reduction in Charpy USE for its surveillance weld and plate are less than or equal to the values projected using the methodology in RG 1.99, Revision 2.

In its response to RAI 4.6-3 and in Section E of the LRA, the applicant provided plant-specific information necessary to demonstrate that the Plant Hatch beltline materials meet the criteria specified in the report. The applicant indicates that the predicted reduction in Charpy USE at 54 EFPY for the limiting plates in Units 1 and 2 is 19 percent and 15 percent, respectively. The predicted reduction in Charpy USE at 54 EFPY for the limiting welds in Units 1 and 2 is 33 percent and 24 percent, respectively. The applicant indicates that the percent reduction in Charpy USE for its surveillance weld and plate is less than the values projected using the methodology in RG 1.99, Revision 2. The staff has reviewed the information provided by the applicant, and has determined that the percent reduction in Charpy USE for the beltline materials and the surveillance materials meet the criteria specified in EPRI TR-113596. In addition, the staff has also determined that the materials and surveillance data reported by the applicant are consistent with data contained in the Reactor Vessel Integrity Database (RVID). The RVID is a database maintained by the staff, which contains a summary of all of the relevant materials data submitted by all applicants in their evaluations of reactor vessel integrity. Since the Plant Hatch beltline material and surveillance weld and plate meet the specified criteria, the Plant Hatch beltline materials will meet the margins of safety against fracture equivalent to those required by Appendix G to the ASME Code and, therefore, will meet the Charpy USE requirements of Appendix G to 10 CFR Part 50 at 54 EFPY.

ART

The staff evaluated the P-T limit curves prepared on the basis of NRC regulations and guidance, including Appendix G to 10 CFR Part 50; GL 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations"; GL 92-01, "Reactor Vessel Structural Integrity," Revision 1; GL 92-01, Revision 1, Supplement 1; and RG 1.99, Rev. 2. GL 88-11 advised applicants that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. GL 92-01, Rev. 1, requested that applicants submit their RPV data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that applicants provide and assess data from other applicants that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves. Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code.

SRP Section 5.3.2 provides an acceptable method to determine the P-T limit curves for ferritic materials in the beltline of the RPV on the basis of the linear elastic fracture mechanics (LEFM) methodology specified in Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methodology specified in Appendix G postulates the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-quarter thickness (1/4T) of the RPV beltline and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T curves are the 1/4T and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside and outside surface defects, respectively.

The Appendix G to the ASME Code methodology requires that applicants determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material, and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or generic value, and whether the chemistry factor (CF) was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

Tables 3-1 and 3-2 in Enclosure 3 to Section E contains the applicant's evaluation of the ART for all RPV beltline materials in Plant Hatch Units 1 and 2 at 54 EFPY.

The material with the highest ART at 54 EFPY in the RPV beltlines of Unit 1 is plate G-4804-2. This plate contains 0.13 percent copper and 0.70 percent nickel, which, according to RG 1.99, Revision 2, corresponds to a chemistry factor of 93.5. This chemistry factor was increased by a factor of 2.62 on the basis of the test results from the reactor vessel materials surveillance program. This results in a chemistry factor for this plate of 245 (93.5 x 2.62). The neutron fluence at the 1/4T location for this plate at 54 EFPY is 2.51E18 n/cm^2, which corresponds to a fluence factor of 0.625. The product of this fluence factor and a chemistry factor of 245 results in a ΔRT_{NDT} at 54 EFPY of 153.2 °F. Since the initial RT_{NDT} for this plate is -20° F and the margin term is 34 °F, the ART for this plate at 54 EFPY is 167.2 °F.

The material with the highest ART at 54 EFPY in the RPV beltline of Unit 2 is plate G-6603-2. This plate contains 0.083 percent copper and 0.58 percent nickel, which, according to RG 1.99, Revision 2, corresponds to a chemistry factor of 51. This chemistry factor was determined using Table 2 of RG 1.99, Revision 2, since no surveillance data exist for this material. The neutron fluence at the 1/4T location for this plate at 54 EFPY is 1.67E18 n/cm^2, which corresponds to a fluence factor of 0.527. The product of this fluence factor and a chemistry factor of 51 results in a ΔRT_{NDT} at 54 EFPY of 26.9° F. Since the initial RT_{NDT} for this plate is 24° F and the margin term is 26.9° F, the ART for this plate at 54 EFPY is 77.8° F.

Since the current Plant Hatch P-T limit curves at 54 EFPY meet the requirements of Appendix G to 10 CFR Part 50, the applicant has demonstrated that the Plant Hatch RPV can operate during the license renewal period and satisfy the requirements of Appendix G to 10 CFR Part 50. In the LRA, the applicant provided Section E, which proposed a change to the Unit 1 and 2 technical specifications in support of extended plant operation. Pressure-temperature operating limits predicated on the effects of irradiation on the core beltline up to 32 EFPY were incorporated at the time of submittal of the LRA. Subsequently, the applicant submitted its annual update to the LRA, dated December 15, 2000. In that update, the applicant removed the proposed change to the technical specifications because the applicant has separately requested and received amendments to the technical specifications that incorporate changes to the pressure-temperature operating limits. However, Enclosure 3 to LRA Section E is retained since it supports certain reactor vessel TLAA issues. Those portions of Enclosure 3 that specifically address the pressure-temperature limits are superseded by the separate licensing action taken by the NRC in issuing Amendments 222 and 163 to the Unit 1 and Unit 2 operating licenses, respectively.

Circumferential RPV Weld Inspection

Sections 4.6.3 and A.1.17.1 of the LRA discuss ultrasonic inspection of the Plant Hatch RPV circumferential welds. Section A.1.17.1 of the LRA indicates that Plant Hatch will use an approved technical alternative in lieu of ultrasonic testing of RPV circumferential shell welds. The technical alternative is discussed in the staff's final SER of the BWR Vessel and Internals Project BWRVIP-05 Report, which is contained in a letter dated July 28, 1998 to Carl Terry, BWRVIP Chairman. In that letter, the staff concludes that, since the failure frequency for circumferential welds in BWR plants is significantly below the criteria specified in RG 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," and the core damage frequency (CDF) of any BWR plant, and since continued inspection would result in a negligible decrease in an already acceptably low value, elimination of the ISI for RPV circumferential welds is justified. The staff's letter indicates that BWR applicants may request relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds by demonstrating (1) at the expiration of the license, the circumferential welds satisfy

the limiting conditional failure probability for circumferential welds in the evaluation, and (2) they have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the report. The letter indicated that the requirements for inspection of circumferential RPV welds during an additional 20-year license renewal period will be reassessed, on a plant-specific basis, as part of any BWR license renewal application.

Section A.4.5 of Report BWRVIP-74 indicates that the staff's SER conservatively evaluated BWR RPVs to 64 effective full-power years (EFPY), which is 10 EFPY greater than what is realistically expected for the end of the license renewal period. Since this was a generic analysis, the applicant must provide plant-specific information to demonstrate that the Plant Hatch beltline materials meet the criteria specified in the report.

In response to RAI 4.6-1, the applicant indicates that procedures and training used to limit cold overpressure events during the license renewal period will be the same as those approved by the NRC when Plant Hatch requested that the BWRVIP-05 technical alternative be used for the current term. In addition, the applicant compared the mean RT_{NDT} for Combustion Engineering fabricated welds from the staff's SER dated July 28, 1998, to the mean RT_{NDT} of the circumferential welds in Plant Hatch Units 1 and 2 at 54 EFPY. The mean RT_{NDT} values in the staff's SER were determined for the limiting BWR RPVs that were fabricated by Combustion Engineering, Babcock and Wilcox, and Chicago Bridge and Iron. Since the Plant Hatch RPVs were fabricated by Combustion Engineering, the results from the staff's SER are applicable to Plant Hatch. However, the mean RT_{NDT} values projected for the circumferential welds at Plant Hatch were calculated using the neutron fluence at the 1/4T location, and included a margin term. The mean RT_{NDT} in the staff's SER was determined using the neutron fluence at the clad/weld metal interface, and did not include a margin term. In a letter dated January 31, 2001, the applicant revised its analysis on the basis of the projected neutron fluence at the clad/weld interface, and did not include a margin term when calculating the mean RT_{NDT}. The mean RT_{NDT} of the circumferential welds in Hatch at 54 EFPY is less than the values for Combustion Engineering vessel (using Combustion Engineering Owners Group chemistries) at 32 EFPY and 64 EFPY, which indicates that the Plant Hatch circumferential welds will be less embrittled than the Combustion Engineering vessel in the NRC staff analysis at 32 EFPY and 64 EFPY. The staff SER indicates that the conditional failure probabilities for the Combustion Engineering vessel at 32 EFPY and 64 EFPY were 6.34E-5 and 4.38.34E-4, respectively. Since the Hatch circumferential welds will be less embrittled than the Combustion Engineering vessel analyzed in the staff's SER, the conditional failure probability for the Hatch RPVs will be less than the values specified in the staff's SER for circumferential welds. Therefore, the applicant has demonstrated compliance with the criteria in the letter dated July 28, 1998, to Carl Terry, and has justified relief from the inservice inspection requirements of 10 CFR 50.55a(g) for volumetric examination of circumferential RPV welds during the license renewal period.

Axially Oriented RPV Welds

In its letter dated July 28, 1998, to Carl Terry, BWRVIP Chairman, the staff also identified a concern about the failure frequency of axially oriented welds in BWR RPVs. In its response to this concern, the BWRVIP provided evaluations of axial weld failure frequency in letters dated December 15, 1998 and November 12, 1999. The staff's evaluation of these analyses is contained in a letter dated March 7, 2000, to Carl Terry. The SER that is enclosed in that letter indicates that the RPV failure frequency as a result of the failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is below 5×10^{-6} per reactor year, given the assumptions regarding flaw density, distribution, and location described in the SER. Since the results apply only for the initial 40-year

license period of BWR plants, applicants for license renewal must provide plant-specific information applicable to 60 years of operation.

The BWRVIP identified Clinton and Pilgrim as the reactor vessels with the highest mean RT_{NDT} in the BWR fleet. The staff confirmed this conclusion in its SER by comparing the information contained in the BWRVIP analysis and the information contained the RVID for all BWR RPV axial welds. The staff performed analyses of the Clinton and Pilgrim plants. The results from the staff's calculations are provided in Table 1. The staff's calculations used the basic input information for Pilgrim, with three different assumptions for the initial RT_{NDT}. The calculations of the actual Pilgrim condition used the docketed initial RT_{NDT} of -48°F and a mean RT_{NDT} of 68°F. A second calculation, listed as "Mod 1" in Table 1, is consistent with the BWRVIP calculations, with an initial RT_{NDT} of 0°F and a mean RT_{NDT} of 116°F. A third calculation, with an initial RT_{NDT} of -2°F and a mean RT_{NDT} of 114°F, was chosen to identify the mean value of RT_{NDT} required to provide a result that closely matches the RPV failure frequency of 5×10^{-6} per reactor-year.

Table 1: Comparison of Results from Staff and BWRVIP

Plant	Initial RT_{NDT} (°F)	Mean RT_{NDT} (°F)	Vessel Failure Freq.	
			Staff	BWRVIP
Clinton	-30	91	2.73E-6	1.52E-6
Pilgrim	-48	68	2.24E-7	-------
Mod 1 *	0	116	5.51E-6	1.55E-6
Mod 2 **	-2	114	5.02E-6	-------

* A variant of Pilgrim input data, with initial RT_{NDT} = 0°F
** A variant of Pilgrim input data, with initial RT_{NDT} = -2°F

The applicant provided plant-specific information in response to RAI 4.6-2 to demonstrate that the Plant Hatch beltline materials meet the criteria specified in the SER. The mean RT_{NDT} for the Plant Hatch axial welds were not compared to the mean RT_{NDT} in Table 1. Instead, the mean RT_{NDT} was compared to the mean RT_{NDT} for axial welds in the staff's SER dated July 28, 1998. The SER in the letter dated March 7, 2000, supersedes the analysis in the letter dated July 28, 1998. In a letter dated January 31, 2001, the applicant revised its analysis to compare the mean RT_{NDT} for the Plant Hatch axial welds to the mean RT_{NDT} for Pilgrim Mod 2 in Table 1, above. The mean RT_{NDT} of the axial welds at Hatch at 54 EFPY was less than 114°F for both units. This value is less than the value for Pilgrim Mod 2 in Table 1, which indicates that the Hatch axial welds at 54 EFPY will be less embrittled than the axial welds for the Pilgrim Mod 2 analysis performed by the staff in its letter dated March 7, 2000. Since the Plant Hatch axial welds will be less embrittled than the axial welds for the Pilgrim Mod 2 analysis performed by the staff in its letter dated March 7, 2000, the conditional failure probability for the Plant Hatch RPVs will be less than 5×10^{-6} per reactor-year at 54 EFPY. Therefore, the applicant has demonstrated compliance with the criteria in the staff's letter dated March 7, 2000.

Neutron Fluence of the RPV

The Charpy USE, ART, circumferential weld, and axial weld RPV integrity evaluations are all dependent upon the neutron fluence. The neutron fluences for the Plant Hatch units were calculated using the General Electric methodology documented in surveillance capsule reports GE-NE-B1100691-01R1 (March 1997) and SASR 90-104 (May 1991). These neutron fluences were determined by taking the fluence at 32 EFPY associated with the approved extended power uprate, and adding to it the fluence that would accumulate during an additional 22 EFPY of operation at the flux associated with the extended power uprate conditions. The extended power uprate was approved in a letter to HL Sumner, Jr., dated October 22, 1998; therefore, the neutron fluences documented in the LRA are acceptable at this time.

4.6.3 Conclusions

The staff has reviewed the information in Section 4.6, "Reactor Vessel TLAAs" of the LRA and the applicant's responses to the staff's RAIs. On the basis of this review, the staff concludes that the applicant has adequately evaluated the reactor vessel TLAA as required by 10 CFR 54.21(c)(1).

4.7 Main Steam Isolation Valves Operating Cycles

4.7.1 Introduction

The applicant described its evaluation related to main steam isolation valve operating cycles in Section 4.7, "Main Steam Isolation Valves Operating Cycles," of the LRA. The staff reviewed this section of the LRA to determine whether the applicant has adequately evaluated the TLAA as required by 10 CFR 54.21(c).

4.7.2 Summary of Technical Information in Application

The Plant Hatch UFSARs contain statements with regard to the design of the MSIVs for the current license term. Section 5.5.5.1of the Unit 2 UFSAR, states the following (with a similar reference in Section 4.6.3 of the Unit 1 UFSAR):

> "The design objective for the valve is a minimum 40-year service at the specified operating conditions. Operating cycles are estimated to be 100 cycles per year during the first year and 50 cycles per year thereafter."

The applicant further stated that the UFSAR statement refers to mechanical cycles of the valve. Cycling of the valve will lead to wear of the valve disc and valve seat. The wear will accumulate over time, (2050 cycles are assumed in the UFSAR statement for 40 years.) The statement, therefore, meets the criteria of a TLAA. However, this type of wear as a result of valve operation will lead to performance degradation that can be discovered through normal leakage monitoring testing. Excessive leakage would lead to refurbishment or repair of the valve set and disc, as necessary. Once the maintenance is performed, the service life of the valve would be restored. Since the aging effects can be readily discovered through normal Technical Specification surveillance testing and repairable maintenance, the TLAA is demonstrated through Criterion (iii) of 10 CFR 54.21(c)(1).

4.7.3 Staff Evaluation

As described above, the applicant dispositioned this TLAA through Criterion (iii) of 10 CFR 54.21(c)(1). Under this disposition option, the applicant should demonstrate that the effects of aging on the components' intended functions will be adequately managed in a manner that is consistent with the CLB throughout the period of extended operation. In addition, the FSAR Supplement for the facility should contain a summary description of the programs and activities for managing the effects of aging and the evaluation of the TLAA throughout the period of extended operation.

In RAI 4.7-1, dated July 28, 2000, the staff requested that the applicant provide information as described in 10 CFR 54.21(c)(1)(iii). The applicant responded to this RAI in its letters dated October 10, 2000, and January 31, 2001. The applicant stated that at the time of the LRA submittal, GE had been unable to fully determine the basis for the MSIV cycles in the UFSAR. Therefore, as a conservative measure, the applicant identified the MSIV cycles in the UFSAR as a TLAA. Since that time, GE has determined that the number is derived from a specification, not from a calculation or analysis, as discussed in the Rule. On the basis of this confirmation from GE, the applicant has now determined that the MSIV cycles do not constitute a TLAA. The applicant also noted that, outside the scope of license renewal, the MSIVs are extensively tested as part of existing Technical Specification requirements because the valves are within the purview of that rule, and are being maintained in a manner that is consistent with the requirements of the maintenance rule. The applicant noted that the MSIVs have extensive testing programs that implement containment isolation testing and valve stroking requirements contained in Technical Specification 3.6.1.3. There are also inspection procedures to address the wear of the stellite faces. The MSIVs are periodically disassembled and refurbished. The solenoid valves and limit switches on the valves are also routinely replaced or completely refurbished to address environmental qualification requirements. In addition, there are other repetitive tasks, such as replacing the actuator hydraulic fluid every 54 months, and inspecting the wiring every 36 months. In addition, the applicant stated that because these valves are periodically tested and refurbished, as necessary, GE has indicated that it is appropriate to restore the valve service life when valve internals are refurbished.

On the basis of this supporting information, even if the assumption were made that the UFSAR text constituted a *de facto* TLAA that is not directly supported by a calculation or analysis, the periodic restoration of the valve service life results in the supposed TLAA failing the criterion that the calculation or analysis must be relevant to making a safety-related determination. The applicant further noted that although the MSIV cycles do not constitute a TLAA as presented in the LRA, the MSIV valve bodies are within the scope of license renewal and are subject to an AMR.

4.7.4 Conclusion

The staff has reviewed the information in Section 4.7, "Main Steam Isolation Valves Operating Cycles" of the LRA and the applicant's responses to the staff's RAI. On the basis of this review, the staff concludes that the applicant's responses are reasonable and sufficient for concluding that MSIV operating cycles do not constitute a TLAA and, therefore, are acceptable.

5 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

During the 481st meeting of the Advisory Committee on Reactor Safeguards (ACRS) on April 5, 2001, the ACRS reviewed the NRC staff's safety evaluation report (SER) related to the license renewal application (LRA) for the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (Plant Hatch). The ACRS Subcommittee on Plant License Renewal initially reviewed the SER prior to its meeting with the NRC staff and the applicant on March 28, 2001, and presented its findings during the April 5, 2001 ACRS Full committee meeting. On April 16, 2001, the ACRS Full committee issued an interim letter on its review of the Plant Hatch license renewal SER with open items.

The staff issued its final SER related to the LRA for Plant Hatch, with the resolution of the open items, on October 5, 2001. The staff briefed the ACRS License Renewal Subcommittee on October 25, 2001. During the 487th meeting of the ACRS Full committee on November 8, 2001, the ACRS completed its review of the Plant Hatch LRA and the staff's SER, and documented its findings in a letter dated November 16, 2001. A copy of that letter is provided.

November 16, 2001

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2

During the 487th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 2001, we completed our review of the Southern Nuclear Operating Company's (SNC's) application for license renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2, and the related final Safety Evaluation Report (SER). We issued an interim letter concerning this application and the SER with open items on April 16, 2001, and our Plant License Renewal Subcommittee held discussions with representatives of the staff and SNC on October 25, 2001. We also had the benefit of the documents referenced.

Conclusions and Recommendations

1. The SNC application for renewal of the operating licenses for Hatch, Units 1 and 2, should be approved.

2. The programs instituted to manage aging-related degradation are appropriate and provide reasonable assurance that Hatch, Units 1 and 2, can be operated safely in accordance with their licensing bases for the period of extended operation without undue risk to the health and safety of the public.

3. The staff has performed a comprehensive review of SNC's application. The open items identified in the February 2001 draft SER have been resolved satisfactorily.

4. The SER clarifies staff positions on non-safety-related seismic II-over-I piping systems, long-lived passive components of skid-mounted complex assemblies, fan housings, and damper frames. These clarifications provide significant guidance that could prevent these issues from becoming open items in future applications. They should be incorporated into the generic license renewal guidance documents.

Background and Discussion

This report fulfills the requirement of 10 CFR 54.25 that the ACRS review and report on license renewal applications. SNC requested renewal of the operating licenses for Hatch, Units 1 and 2, for a period of 20 years beyond the current license terms, which expire on August 6, 2014, for Unit 1, and June 13, 2018, for Unit 2. The final SER documents the results of the staff's review of information submitted by SNC, including those commitments that were necessary to resolve open items identified by the staff in its February 2001 draft SER. The staff's review included the verification of the completeness of structures, systems, and components (SSCs) identified in the application, the validation of the integrated plant assessment process, the identification of the possible aging effects associated with each passive long-lived component, and the verification of the adequacy of the aging management programs. The staff also conducted site inspections to verify the adequacy of the implementation of the methodology described in the application.

As noted in our April 16, 2001 interim letter, the SNC's approach to identifying SSCs that are within the scope of the License Renewal Rule is function-based, rather than the system-based approach used in previous applications. This approach was adequate, but made it difficult for the reviewers to ascertain which SSCs were in scope and which were not. The staff's review relied heavily on supporting documents located at the site and on requests for additional information. In addition, the staff performed a "walk-through" of the process for three systems that are within scope. On the basis of its extensive review, the staff identified some additional components that the applicant should have included within the scope of license renewal, and classified them as open items. These open items have been resolved by including the additional components in scope. We concur with the staff that the applicant has now properly identified SSCs requiring an aging management review.

Components brought into scope through the resolution of open items include non-safety-related seismic II-over-I piping systems, long-lived passive components of skid-mounted complex assemblies, fan housings, and damper frames. The inclusion of these components was contested in previous license renewal applications. The issue of seismic II-over-I piping is an open item in an application that is currently under review. The Hatch SER includes effective clarifications of why these components need to be included within scope. The guidance provided by these clarifications could prevent these issues from becoming open items in future applications. Consequently, these clarifications should be incorporated into the generic license renewal guidance documents.

SNC has conducted a comprehensive aging management review of SSCs that are within scope. Aging effects were identified on the basis of component material, operating environment, and operating stresses using plant-specific and industry-wide operating experience. Topical reports developed by the Boiling Water Reactor Vessel and Internals Project (BWRVIP) were also used to identify aging effects and to develop aging management programs that support the Hatch application. We reviewed a number of BWRVIP topical reports and commented on their effectiveness in supporting license renewal in our April 16, 2001 letter.

Appendix A to the Hatch application describes 17 existing programs, 5 modified programs, and 7 new programs that SNC has implemented to manage aging effects during the period of extended operation. The resolution of open items has resulted in added commitments to these programs, including a one-time inspection of plant service water piping in the diesel generator building and a one-time inspection of small-bore butt-welded stainless steel piping.

One of the added commitments resulting from resolution of open items involves periodic testing of fire-protection system sprinkler heads that are within the scope of license renewal. SNC had proposed a one-time test of such sprinkler heads at or before the start of the period of extended operation. The staff did not agree with the one-time test, because the design life (50 years) of the sprinkler heads does not cover the period of extended operation. As recommended by the staff, SNC has committed to perform the sprinkler head tests as specified in the National Fire Protection Association (NFPA) Standard 25, Section 2.3.3.1, "Sprinklers." The application of this Standard will result in periodic testing of the sprinkler heads at 10-year intervals, with the first test taking place during the third year of the renewal period. This program is acceptable because it confirms the effectiveness of the periodic inspections to which the sprinkler heads are subjected and ensures testing of the sprinkler heads early in the renewal period.

The staff requested that SNC perform a one-time inspection of the four buried emergency diesel generator (EDG) fuel oil storage tanks. SNC responded by performing visual inspections and ultrasonic testing of one of the four tanks. Ultrasonic testing of 144 locations along the lower shell of the tank indicated that there was no thinning of the wall. Visual inspections of the internal surface revealed very little corrosion. SNC and the staff concluded that the one-time inspection demonstrated that loss of material of the diesel fuel oil storage tanks was not an aging effect requiring management during the period of extended operation.

We also considered the possibility that the external coating of a tank could be damaged at some location during installation and result in localized fuel oil leakage. Such damage would be of concern during the current license term and, thus, would not be specific to the period of extended operation. The safety consequences would not be significant because the potential leakage would not cause substantial depletion of the fuel oil inventory before it would be detected. We concur with the staff's determination that loss of material of the diesel fuel oil storage tanks is not an aging effect requiring management during the period of extended operation.

Jet pump assemblies and fuel supports contain cast austenitic stainless steel (CASS) components that are within the scope of license renewal. These components may be exposed to neutron fluence levels that would make them susceptible to neutron irradiation embrittlement and loss of fracture toughness. Since neutron embrittlement becomes a concern when cracks are present in the components, the staff requested that SNC propose a one-time inspection of the jet pump assemblies and fuel supports to confirm that these CASS components have not experienced cracking. Following this request, the staff recognized that cracking of CASS components has not been observed to date. Furthermore, BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," requires inspections of jet pump assembly welds that are generally believed to be more susceptible to cracking than the CASS components and, therefore, provide a leading indicator for inspection of CASS components. SNC has committed to perform the weld inspection required by BWRVIP-41. In addition, the BWRVIP and the NRC's Office of Nuclear Regulatory Research plan to conduct confirmatory research to determine the effects of high levels of neutron fluence on BWR internals. SNC has committed to implement any requirements resulting from this research. Given the above, the staff concluded that the requested one-time inspection is not warranted at this time. We agree with the staff's conclusion.

Time-limited aging analyses (TLAA) have shown that neutron irradiation embrittlement during the extended period of operation will have no significant impact on the integrity of the Hatch

reactor vessels. At the end of the renewal period, the vessels will still have margin over applicable regulatory limits. In order to monitor time-dependent parameters used in the TLAA, SNC plans to implement the provisions of the integrated surveillance program (ISP) described in BWRVIP-78, "BWR integrated surveillance program plan," and BWRVIP-86, "BWR integrated surveillance program implementation plan. " Since these topical reports have not yet been approved by the staff, SNC committed to implement either a staff-approved ISP or a plant-specific program that meets specific staff requirements on periodic removal of capsules to monitor neutron fluence and the impact of irradiation on the reactor vessels. SNC committed to provide the staff with program details prior to the period of extended operation. The staff made this commitment a license condition.

The staff has performed a comprehensive review of SNC's application. The applicant and the staff have identified plausible aging effects associated with passive and long-lived components. Adequate programs have been established to manage the effects of aging so that Hatch, Units 1 and 2, can be operated safely in accordance with their current licensing bases for the period of extended operation.

Sincerely,

George E. Apostolakis
Chairman

References:

1. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2," issued October 2001.

2. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of the Edwin I. Hatch Nuclear Plant, Units 1 and 2," issued February 2001.

3. Letter dated February 29, 2000, from H. L. Sumner, SNC, to the U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Application for Renewed Operating Licenses."

4. Letter dated April 16, 2001, from George E. Apostolakis, Chairman ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Interim Letter Related to the License Renewal of Edwin I. Hatch Nuclear Station, Units 1 and 2.

5. Topical Report BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines," October 1997.

6. Topical Report BWRVIP-78, "BWR Integrated Surveillance Program - Unirradiated Charpy Reference Curves for Surveillance Material," December 1999.

7. Topical Report BWRVIP-86, "BWR Vessel and Internals Project, BWR Integrated Surveillance Program Implementation Plan."

6 CONCLUSIONS

The staff reviewed the Edwin I. Hatch Nuclear Plant, Units 1 and 2, license renewal application in accordance with Commission regulations and the NRC draft "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," dated September 1997. In 10 CFR 54.29, the staff identifies the standards for issuance of a renewed license.

On the basis of its evaluation of the application as discussed above, the staff has determined that the requirements of 10 CFR 54.29(a) have been met.

The staff notes that the requirements of subpart A of 10 CFR Part 51 are documented in the final plant-specific supplement to the Generic Environmental Impact Statement, dated May, 2001.

APPENDIX A

CHRONOLOGY

This appendix contains a chronological listing of routine licensing correspondence between the U.S. Nuclear Regulatory Commission (NRC) staff and Southern Nuclear Operating Company, Inc. (the applicant) regarding the staff's review of the Edwin I. Hatch Nuclear Plant, Units 1 and 2, Hatch, application for license renewal (Docket Nos. 50-321 and 50-366.)

October 27, 1997	In a letter (signed by H. L. Sumner) SNC indicated its intention to proceed forward with preparing a license renewal application for Plant Hatch Units 1 and 2 and requested a waiver of review fees (ACN 9711040157)
January 15, 1998	In a letter (signed by H. L. Sumner) SNC informed NRC of its plans for product submittals for 1998 (ACN 9801230066)
April 13, 1998	In a letter (signed by H. L. Sumner) SNC informed NRC of its support for Baltimore Gas & Electric Company's License Renewal Application for Calvert Cliffs Nuclear Power Plant.
April 13, 1998	In a letter (signed by H. L. Sumner) SNC submitted its License Renewal Process Methodology Document for Plant Hatch (ACN 9804220149)
May 3, 1998	In a letter (signed by S. Collins) NRC acknowledged SNC's interest in license renewal for Plant Hatch (ACN 9805060036)
January 7, 1999	In a letter (signed by H. L. Sumner) SNC submitted the Hatch, Intake Structure Licensing Report as an example of the technical content and level of detail that Plant Hatch is planning for its application for license renewal (ACN 9901130111)
January 25, 1999	In a letter (signed by W. G. Hairston) SNC informing the NRC of its support for the Commission's recent initiatives to streamline the hearing process (ACN 9903160142)
May 14, 1999	In a letter (signed by H. L. Sumner) SNC submitted the attached Recirculation System Pressure Boundary Licensing Report to provide the NRC with an example of the technical content and level of detail that Plant Hatch is planning for its application for License Renewal (ML003704042)

November 12, 1999	In a letter (signed by H. L. Sumner) SNC requested exemption from 10 CFR 50.30(a)(2), 51.55(a), and 2.101(a)(3), and requested exception to 10 CFR 50,4(b) and 50.4(c): written submittal requirements (ML993270222)
January 24, 2000	In a letter (signed by B. Shelton) NRC responded to SNC's request for exemption from 10 CFR 50.30(a)(2), 51.55(a), and 2.101(a)(3), and request for exception to 10 CFR 50.4(b) and 50.4(c): written submittal requirements (ML003677239)
February 29, 2000	In a letter (signed by H. L. Sumner) SNC submitted its License Renewal Application (LRA) for Edwin I. Hatch Nuclear Plant, Units 1 and 2, (Hatch) (ML003688151)
February 29, 2000	In a letter (signed by H. L. Sumner) SNC submitted its associated evaluation boundary drawings for the Plant Hatch Application for Renewed Operating Licenses (ML003688222)
March 3, 2000	In a letter (signed by C. Grimes) NRC informed SNC of the receipt of the Edwin I. Hatch, Units 1 and 2, LRA and Assignment of a Project Manager (ML003688811)
March 24, 2000	In a letter (signed by C. Grimes) NRC informed SNC of the determination of acceptability and sufficiency for docketing, proposed review schedule, and opportunity for a hearing regarding an application from SNC for renewal of the operating licenses for Units 1 and 2 of the Edwin I. Hatch Nuclear Plant (ML003695605)
April 4, 2000	In a letter (signed by D. Matthews) NRC informed SNC of the preparation of a notice of intent that advises the public that the NRC intends to gather information necessary to prepare a plant-specific supplement to the Commission's "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," (NUREG-1437) in support of the review of the application for the renewal of the Hatch operating license.
April 4, 2000	In an electronic correspondence (signed by R. Baker) SNC provided the expanded matrix of programs/activities and commodity groups with a "system" column added.
April 6, 2000	In a memorandum (signed by W. Burton) NRC issued a public meeting notice to the stakeholders and the public and informed them of a meeting to be held on April 12, 2000, with SNC to familiarize the staff reviewers with Hatch scoping methodology and boundary drawings (ML0037000062)
April 12, 2000	Federal Register Notice announcing environmental scoping meeting

April 14, 2000	In a memorandum (signed by W. Burton) NRC provided SNC with a summary of the April 19, 2000, teleconference with SNC regarding aging management program A.3.7 of the Hatch LRA, "Torus Submerged Components Inspection Program."
April 28, 2000	In a memorandum (signed by R. K. Anand) NRC issued a public meeting notice to stakeholders and the public and informed them of a meeting to be held on May 8, 2000, with SNC to discuss progress of aging management program review of SNC's LRA for Hatch.
May 1, 2000	In a memorandum (signed by W. Burton) NRC provided SNC with a summary of the working meeting on April 12, 2000, with SNC, regarding scoping review for Hatch LRA.
May 4, 2000	In a memorandum (signed by S. Hoffman) NRC issued a public meeting notice to stakeholders and the public and informed them on a meeting to be held on May 17, 2000, between the NRC's License Renewal Steering Committee with the Nuclear Energy Institute's (NEI's) License Renewal Working Group to discuss NRC and industry generic license renewal activities
May 23, 2000	In an electronic correspondence (signed by R. Baker) SNC provided the requested recent board changes in Oglethorpe Power Corporation.
May 24, 2000	Changes to Oglethorpe Power's principal officers (ML003718346)
May 24, 2000	In an electronic correspondence (signed by R. Baker) SNC submitted a database sort of unctions on a system-by-system basis and another sort of present systems on a function-by-function basis (ML003718384)
May 30, 2000	In a letter (signed by J. H. Wilson) NRC requested additional information related to the staff's review of Severe Accident Mitigation Alternatives for Hatch (ML003719228)
May 31, 2000	In a letter (signed by H. L. Sumner) SNC provided additional information supporting license renewal environmental report and submitted a copy of a matrix developed during SNC's review of Category 1 items for new and significant information (ML003719941)
June 1, 2000	In a memorandum (signed by R. Prato and W. Burton) NRC provided SNC with a summary of the May 17, 2000, meeting with Entergy and SNC regarding license renewal activities for Arkansas Nuclear One - Unit 1 (ANO-1) and Hatch (ML003720297)

June 4, 2000	Supplement to May 10, 2000 testimony and additional statement (ML003722562)
June 16, 2000	In an electronic correspondence (signed by R. Baker) SNC provided a revised function to system matrix to replace the version provided on May 24, 2000.
June 20, 2000	In a letter (signed by C. A. Casto) NRC informed SNC of the license renewal inspection schedule for Hatch.
June 20, 2000	In an electronic correspondence (signed by R. Baker) SNC submitted a matrix mapping the cracking aging mechanisms discussed in the various C.2 commodity groups.
June 23, 2000	In a letter (signed by J. H. Wilson) NRC requested additional information related to the staff's review of the License Renewal Environmental Report for Hatch (ML003726207)
June 27, 2000	In a letter (signed by W. Burton) NRC informed SNC of the schedule revision for the review of the Hatch LRA (ML003726800)
June 27, 2000	In a letter (signed by L. N. Olshan) NRC requested additional information concerning the Liquid and Gaseous Radwaste System at Hatch (ML003727407)
July 14, 2000	In a letter (signed by W. Burton) NRC requested additional information (RAI) on LRA Sections 2.1, 2.2, 2.3.1, (SER Section 2.3.2), 2.3.2, 2.3.3, 2.3.4, 2.3.5, 2.4, 2.5, 3.2.3 (SER Section 3.3), and 3.2.5 (SER Section 3.5) (ML003732558)
July 26, 2000	In a letter (signed by H. L. Sumner) SNC provided its response to the NRC RAIs related to the review of severe accident mitigation alternatives for Hatch.
July 26, 2000	In an electronic correspondence (signed by H. L. Sumner) SNC provided its response to the NRC RAIs concerning the Liquid and Gaseous Radwaste System at Hatch (ML003736984)
July 28, 2000	In a letter (signed by W. Burton) NRC request SNC to provide additional information (RAI) on LRA Sections 2.3.3, 2.3.4 (ML003736523)
August 11, 2000	In a memorandum (signed by W. Burton) NRC issued a non-public meeting notice to stakeholders informed them on a meeting to be held on August 23, 2000, to review samples of Hatch environmental qualification calculations to verify calculation methods as applies in the Hatch LRA (ML003740331)

August 11, 2000	In a letter (signed by H. L. Sumner) SNC provided its response to the NRC RAIs on the renewal environmental report of the Hatch LRA requested on June 23, 2000.
August 21, 2000	In a letter (signed by H. L. Sumner) SNC provided its response to the NRC RAIs on the scoping and screening (Section 2 of the LRA) and aging management issues (Section 3 or 4 of the LRA) by providing a proposed schedule for response to these requests for additional information.
August 23, 2000	In a memorandum (signed by W. Burton) NRC provided SNC with a summary of the August 23, 2000, meeting with SNC regarding environmental qualification calculations for Hatch.
August 29, 2000	In an electronic correspondence (signed by R. Baker) SNC submitted drawing HL-16040 in response to RAI2.3.4-CBHVAC-4.
August 29, 2000	In a memorandum (signed by B. Boger and C. Casto) NRC informed SNC of the final Hatch License Renewal Inspection Plan (ML003745955)
August 29, 2000	In a letter (signed by H. L. Sumner) SNC provided its response to the NRC RAIs on the scoping and screening (Section 2) requested on July 14, 2000, and July 28, 2000 (ML003746406)
August 31, 2000	In a letter (signed by H. L. Sumner) SNC provided clarification on the requested additional information (RAI) related to the review of severe accident mitigation alternative dated May 30, 2000.
September 25, 2000	In an electronic correspondence (signed by W. Burton) NRC provided SNC with a correction to RAI 3.1.5-6 and a summary of the staff position on complex assemblies.
October 1, 2000	In an electronic correspondence (signed by R. Baker) SNC provide NRC with the Hatch License Renewal Scoping Inspection follow-up items that remained outstanding following the technical debrief on September 15, 2000.
October 6, 2000	In an electronic correspondence (signed by W. Burton) NRC provided SNC with the revision to the first part of RAI 3.1.18-10.
October 10, 2000	In a letter (signed by H. L. Sumner) SNC provided its response to the NRC remaining RAIs on aging management programs requested on June 23, 2000, that were not covered in the SNC response dated August 29, 2000 (ML003759631)
October 13, 2000	In an electronic correspondence (signed by R. Baker) SNC provided NRC with the scoping/screening RAI follow-ups on fire protection.

October 19, 2000	In an electronic correspondence (signed by W. Burton) NRC provided SNC with a revision of the August 23, 2000, meeting summary for the EQ calculations.
October 19, 2000	In an electronic correspondence (signed by W. Burton) NRC provided SNC with a summary of the September 13, 2000, and September 28, 2000, telecon related to fire protection.
October 20, 2000	In an electronic correspondence (signed by W. Burton) NRC provided SNC with a summary of the September 13, 2000, and September 28, 2000, telecon related to HR, P&I, AD, COND, DPS, EDG, IA, and EHC.
October 20, 2000	In an electronic correspondence (signed by W. Burton) NRC provided SNC with a summary of the September 13, 2000, and September 28, 2000, telecon related to RC, SLC, RHR, and CRD.
October 20, 2000	In an electronic correspondence (signed by W. Burton) NRC provided SNC with a summary of the June 27, 2000, telecon related to plant service water and traveling water screens/trash racks.
October 20, 2000	In an electronic correspondence (signed by W. Burton) NRC provided SNC with a summary of the June 29, 2000, telecon related to RCS, SLC, and BWRVIP.
October 25, 2000	In a letter (signed by A. Kugler) NRC requested comment on the draft plant-specific supplement to the "Generic Environmental Impact Statement for License Renewal of Nuclear Plants [GEIS]" (NUREG-1437) regarding Hatch (ML003767639)
November 1, 2000	In a letter (signed by C. Casto) NRC provided SNC with the scoping inspection report of the results of the inspection at the Birmingham, Alabama offices regarding SNC's Hatch LRA (ML003773009)
November 3, 2000	Draft Supplemental Environmental Impact Statement (ML003766660)
November 8, 2000	In an electronic correspondence (signed by W. Burton) NRC provided SNC with a summary of the October 24, 2000, telecon related to RHR heat exchangers and treatment of seismic II/I piping
December 13, 2000	In a letter (signed by H. L. Sumner) SNC requested a partial fee waiver of 40 percent and that the staff take appropriate measures to account for its time so that such a waiver is realized (ML003779256)

December 15, 2000	In a letter (signed by H. L. Sumner) SNC submitted the required amendment (annual update) to the LRA originally submitted February 29, 2000 (ML003781913)
January 5, 2001	In a letter (signed by W. Burton) NRC provided SNC with a draft of open items from the review of the Hatch LRA (ML010050321)
January 16, 2001	In a letter (signed by W. Burton) NRC provided SNC with a schedule revision for the review of the Hatch LRA (ML010170351)
January 31, 2001	Responses to draft open items provided to SNC by letter dated January 5, 2001 (ML010430244)
February 7, 2001	License Renewal Safety Evaluation Report for the Edwin I. Hatch Nuclear Plant (ML01039007)
February 9, 2001	Transmittal of Calculational Summary (Non-Proprietary) (ML010470127)
February 9, 2001	Schedule Revision (ML010430024)
March 19, 2001	Additional Information for GEIS (ML010730246)
April 16, 2001	ACRS Interim Letter on Hatch License Renewal Review (ML011080806)
May 31, 2001	Final Supplemental Environmental Impact Statement (ML011420037)
June 5, 2001	Response to Open Items (ML011620187)
June 14, 2001	Summary of Meeting with Southern Nuclear Corp. on Seismic II/I (ML011660023)
July 26, 2001	Summary of Appeal Meeting (ML012070311)
September 5, 2001	Supplemental Response to Open Items (ML012600021)
September 28, 2001	3rd Inspection Report (ML012730003)
October 5, 2001	Final Safety Evaluation Report (ML012780458/ML012780459)
October 18, 2001	Regional Administrator's Letter (ML012920057)

APPENDIX B

REFERENCES

This appendix contains a listing of references used in the preparation of the Safety Evaluation Report prepared during the review of the license renewal application for Edwin I. Hatch Nuclear Plant Units 1 and 2 under Docket Numbers 50-321 and 50-366).

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ACI 301, "Specifications for Structural Concrete for Buildings."

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ANSI/ANS 56.8-1994, "American National Standard for Containment System Leakage Testing Requirements," 1994.

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ASME Boiler and Pressure Vessel Code, July 1989.

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ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components.

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ASTM A307, "Standard Specification for Carbon Steel Bolts and Steels, 60,000 psi Tensile Strength."

ASTM A325, "Standard Specification for Structural Bolts, Steel, Heat-Treated, 120 ksi and 105 ksi Minimum Tensile Strength."

ASTM A490, “Standard Specification for Heat-Treated Steel Structural Bolts, 150ksi Minimum Tensile Strength.”

ASTM D975-1981, “Standard Specification for Diesel Fuel Oils.”

ASTM, Section 6, Volume 06.02, ”Paints-Products and Applications, Protective Coatings, Pipeline Coatings.”

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AWWA C209, “Cold Applied Tape Coatings for the Exterior of Special Sections, Connections, and Fittings for Steel Water Pipelines,” 1995

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BAW-2270, “Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools,” December 1997.

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BWRVIP-05, “BWR RPV Shell Weld Inspection Recommendations,” September 1995

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BWRVIP-18, “Core Spray Internals Inspection and Flaw Evaluation Guidelines,” July 1996

BWRVIP-26, “Top Guide Inspection and Flaw Evaluation Guidelines,” December 1996

BWRVIP-27, “Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines,” April 1997

BWRVIP-38, “Shroud Support Inspection and Flaw Evaluation Guidelines,” September 1997

BWRVIP-41, “BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines,” October 1997

BWRVIP-47, “BWR Lower Plenum Inspection and Flaw Evaluation Guidelines,” December 1997

BWRVIP-48, “Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines,” March 1998

BWRVIP-60, “Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals,” March 1999

BWRVIP-62, "Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection," December 1998

BWRVIP-74, "BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," September 1999.

BWRVIP-75, "Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (NUREG-0313)," October 1999

BWRVIP-76, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," December 1999.

BWRVIP-78, "BWR Integrated Surveillance Program - Unirradiated Charpy Reference Curves for Surveillance Material," December 1999

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NRC BL 79-01B, "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors," October, 1980.

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

10 CFR Part 50.55a, "Codes and Standards."

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10 CFR Part 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

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

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10 CFR Part 50.Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

10 CFR Part 100, "Reactor Site Criteria."

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Letter from H. L. Sumner (SNC) to NRC, "Reactor Pressure Vessel Shell Welds Examination," dated December 2, 1998.

Letter from B. D. Frew to C. R. Pierce, "Corrosion Allowance for Hatch 1 and 2 Vessel/Piping Systems," dated September 15, 1999, DF 9912, DRF B11-00827.

Letter from C.I. Grimes to D.J. Walters, dated May 19, 2000.

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Letter from C. Grimes (NRC) to H. L. Sumner (SNC), "Request for Additional Information for the Review of the License Renewal Application of Plant Hatch," dated July 28, 2000.

Letter from H. L. Sumner (SNC) to NRC, "Edwin I. Hatch Nuclear Plant - Response To License Renewal Requests for Additional Information," dated August 29, 2000.

Letter from R. L. Emch (NRC) to H. L. Sumner (SNC), "Edwin I. Hatch Nuclear Plant, Units 1 and 2 RE: Evaluation of Relief Requests RR-MC-9 and RR-12: Implementation of Subsections IWE and IWL of ASME Section XI for Containment Inspection," dated October 4, 2000

Letter from H. L. Sumner (SNC) to NRC, "Edwin I. Hatch Nuclear Plant - Response To License Renewal Requests for Additional Information," dated October 10, 2000.

Memorandum from T. Quay (NRC) to C. Grimes (NRC), "Edwin I. Hatch Nuclear Plant License Renewal Application - Scoping and Screening/Corrective Action Process Audit Report," dated October 10, 2000.

Letter from H.L. Sumner to NRC - Transmittal of Responses to License Renewal Draft SER Open Items - June 5, 2001.

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EPRI TR-108705, "BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection," December, 1998 (BWRVIP-62).

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NRC IN 97-72, "Potential for Failure of the Omega Series Sprinkler Heads," September, 1997.

NRC IN 01-10, "Failure of Central Sprinkler Company Model GB Series Fire Sprinkler Heads," June, 2001

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APPENDIX C

ABBREVIATIONS

A/C	air conditioning
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AD	access doors system
AHU	air handling unit
AMP	aging management program
AMR	aging management review
ANL	Argonne National Laboratory
ANSI	American National Standard Institute
ART	adjusted reference temperature
ASME	American Society of Mechnacal Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AWWA	American Water Works Association
BDP	boundary description packages
BOP	balance of plant
BTP	branch technical position
BWR	boiling water reactor
BWROG	BWR Owner's Group
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CAP	corrective actions program
CBHVAC	control building HVAC
CCTLP	component cyclic or transient limit program
CCW	closed cooling water
CDF	core damage frequency
CF	chemistry factor
CFR	*Code of Federal Regulations*
CLB	current licensing basis
CRD	control rod drive
CS	core spray
CST	condensate storage tank
CUF	cumulative usage factor
DBA	design basis accident
DBE	design basis event
DGMA	diesel generator maintenance activities
DOR	Division of Operating Reactors
DWST	demineralized water storage tank
ECCS	emergency core cooling system
ECP	electrochemical potential
EDG	emergency diesel generator

EFPY	effective full power years
EHC	electro-hydraulic control
ELI	equipment location index
EPRI	Electric Power Research Institute
EQ	environmental qualification
EQML	environmental qualification master list
ESF	engineered safety features
FAC	flow accelerated corrosion
FAO	free available oxidant
FHA	fire hazards analysis
FP	fire protection
FR	*Federal Register*
FSAR	final safety analysis report
GALL	generic aging lessons learned
GE	General Electric
GEIS	generic environmental impact statement
GL	generic letter
GSI	generic safety issue
HE/ME	high energy/moderate energy
HELB	high energy line break
HMWPE	high-molecular-weight polyethylene
HNP	Edwin I. Hatch Nuclear Plant
HPCI	high-pressure coolant injection
HVAC	heating, ventilation, and air conditioning
HWC	hydrogen water chemistry
I&E	Inspection and Enforcement
I&E	inspection and evaluation
I/MPO	installation/maintenance procedure manual
IASCC	irradiation assisted stress corrosion cracking
IEEE	Institute of Electrical and Electronic Engineers
IGA	intergranular attack
IGSCC	intergranular stress corrosion cracking
ILRT	integrated leak rate test
IN	information notice
INPO	Institute for Nuclear Power Operations
IPA	integrated plant assessment
ISI	inservice inspection
ISP	integrated surveillance program
LEFM	linear elastic fracture mechanics
LOCA	loss of coolant accident
LOSP	loss of offsite power
LPCI	low-pressure coolant injection
LR	license renewal

LRA	license renewal application
LRT	leak-rate test
LWR	light-water reactor
MC	main condenser
MCR	main control room
MCRE	main control room envelope
MCRECS	main control room environmental control system
MIC	microbiologically-influenced corrosion
MSIV	main steam isolation valve
MSL	main steam line
NDT	nil-ductility transition temperature
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NMCA	noble metal chemical addition
NPAR	nuclear plant aging research
NPDES	national pollutant discharge elimination system
NPS	nominal pipe size
NRC	United States Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
NSOA	nuclear safety operational analysis
NUMARC	Nuclear Management and Resources Council (now NEI)
OSHVAC	outside structures heating, ventilation, and air conditioning system
P-T	pressure-temperature limits
P&I	primary containment purge and inerting
PCCW	primary containment chilled water
PSW	plant service water
PT	liquid penetrant
QA	quality assurance
QDP	qualification data package
RAI	request for additional information
RB	reactor building
RBCCW	reactor building closed cooling water
RBHVAC	reactor building heating, ventilation, and air conditioning system
RCIC	reactor core isolation cooling
RCS	reactor coolant system
RE	refueling equipment system
RG	regulatory Guide
RHR	residual heat removal
RHRSW	RHR service water
RPS	reactor protection system
RPT	recirculating pump trip
RPV	reactor pressure vessel
RRS	reactor recirculation system

RSP	remote shutdown panel
RT	reference temperature
RT	radiographic test
RVID	reactor vessel integrity database
RWCU	reactor water cleanup
SAW	submerged arc weld
SC	structures and components
SCC	stress corrosion cracking
SCFM	standard cubic feet per minute
SE	safety evaluation
SECY	Office of the Secretary of the Commission
SED	system evaluation document
SER	safety evaluation report
SFP	spent fuel pool
SGTS	standby gas treatment system
SLCS	standby liquid control system
SMP	structural monitoring program
SNC	Southern Nuclear Operating Company, Inc.
SOC	statement of consideration
SPCS	steam and power conversion system
SRP	standard review plan
SRP-LR	standard review plan - license renewal
SRV	safety relief valve
SSCs	systems, structures, and components
SSPC	Steel Structures Paint Council
TGSCC	transgranular stress corrosion cracking
TLAA	time-limited aging analysis
TMI	Three Mile Island
TR	technical report
TS	technical specifications
TTA	tolytriazole
TV	tornado vents
TWSPI	treated water systems piping inspection
UFSAR	updated Final Safety Analysis Report
USAS	United States of America Standard
USE	upper shelf energy
UT	ultrasonic test
VFLD	vessel flange leak detection
WG	water gage
XLPE	cross-linked polyethylene

APPENDIX D

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APPENDIX E

REQUESTS FOR ADDITIONAL INFORMATION

RAI	ISSUANCE DATE	RESPONSE DATE	SUBJECT
2.1-SSM-1	July 14, 2000	August 29, 2000	Scoping and Screening Methodology
2.1-SSM-2	July 14, 2000	August 29, 2000	Scoping and Screening Methodology
2.2-SR-1	July 14, 2000	August 29, 2000	Scoping Results
2.2-SR-2	July 14, 2000	August 29, 2000	Scoping Results
2.2-SR-3	July 14, 2000	August 29, 2000	Scoping Results
2.2-SR-4	July 14, 2000	August 29, 2000	Scoping Results
2.3.2-NBS-1	July 14, 2000	August 29, 2000	Nuclear Boiler System
2.3.2-NBS-2	July 14, 2000	August 29, 2000	Nuclear Boiler System
2.3.2-RA-1	July 14, 2000	August 29, 2000	Reactor Assembly System
2.3.2-RA-2	July 14, 2000	August 29, 2000	Reactor Assembly System
2.3.2-RA-3	July 14, 2000	August 29, 2000	Reactor Assembly System
2.3.2-RA-4	July 14, 2000	August 29, 2000	Reactor Assembly System
2.3.3-ESF-1	July 14, 2000	August 29, 2000	Engineered Safety Features
2.3.3-ESF-2	July 14, 2000	August 29, 2000	Engineered Safety Features
2.3.3-HR-1	July 14, 2000	August 29, 2000	Post-LOCA Hydrogen Recombiners

2.3.3-HR-2	July 14, 2000	August 29, 2000	Post-LOCA Hydrogen Recombiners
2.3.3-HR-3	July 14, 2000	August 29, 2000	Post-LOCA Hydrogen Recombiners
2.3.3-HR-4	July 14, 2000	August 29, 2000	Post-LOCA Hydrogen Recombiners
2.3.3-P&I-1	July 14, 2000	August 29, 2000	Primary Containment Purge and Inerting System
2.3.3-P&I-2	July 14, 2000	August 29, 2000	Primary Containment Purge and Inerting System
2.3.3-P&I-3	July 14, 2000	August 29, 2000	Primary Containment Purge and Inerting System
2.3.3-RHR-1	July 14, 2000	August 29, 2000	Residual Heat Removal System
2.3.3-SGTS-1	July 14, 2000	August 29, 2000	Standby Gas Treatment System
2.3.3-SGTS-2	July 14, 2000	August 29, 2000	Standby Gas Treatment System
2.3.3-SGTS-3	July 14, 2000	August 29, 2000	Standby Gas Treatment System
2.3.3-SLCS-1	July 14, 2000	August 29, 2000	Standby Liquid Control System
2.3.3-SLCS-2	July 14, 2000	August 29, 2000	Standby Liquid Control System
2.3.4-AD-1	July 14, 2000	August 29, 2000	Access Doors
2.3.4-CBHVAC-1	July 14, 2000	August 29, 2000	Control Building HVAC
2.3.4-CBHVAC-2	July 14, 2000	August 29, 2000	Control Building HVAC
2.3.4-CBHVAC-3	July 14, 2000	August 29, 2000	Control Building HVAC

2.3.4-CBHVAC-4	July 14, 2000	August 29, 2000	Control Building HVAC
2.3.4-COND-1	July 14, 2000	August 29, 2000	Condensate Transfer and Storage
2.3.4-COND-2	July 14, 2000	August 29, 2000	Condensate Transfer and Storage
2.3.4-COND-3	July 14, 2000	August 29, 2000	Condensate Transfer and Storage
2.3.4-CRD-1	July 14, 2000	August 29, 2000	Control Rod Drive System
2.3.4-DPS-1	July 14, 2000	August 29, 2000	Drywell Pneumatics System
2.3.4-DPS-2	July 14, 2000	August 29, 2000	Drywell Pneumatics System
2.3.4-DPS-3	July 14, 2000	August 29, 2000	Drywell Pneumatics System
2.3.4-EDG-1	July 14, 2000	August 29, 2000	Emergency Diesel Generators System
2.3.4-EDG-2	July 14, 2000	August 29, 2000	Emergency Diesel Generators System
2.3.4-EDG-3	July 14, 2000	August 29, 2000	Emergency Diesel Generators System
2.3.4-FPS-1	July 14, 2000	August 29, 2000	Fire Protection System
2.3.4-FPS-2	July 14, 2000	August 29, 2000	Fire Protection System
2.3.4-FPS-3	July 14, 2000	August 29, 2000	Fire Protection System
2.3.4-FPS-4	July 14, 2000	August 29, 2000	Fire Protection System
2.3.4-FPS-5	July 14, 2000	August 29, 2000	Fire Protection System
2.3.4-FPS-6	July 14, 2000	August 29, 2000	Fire Protection System
2.3.4-FPS-7	July 14, 2000	August 29, 2000	Fire Protection System

2.3.4-FPS-8	July 14, 2000	August 29, 2000	Fire Protection System
2.3.4-FPS-9	July 14, 2000	August 29, 2000	Fire Protection System
2.3.4-FPS-10	July 14, 2000	August 29, 2000	Fire Protection System
2.3.4-IA-1	July 14, 2000	August 29, 2000	Instrument Air System
2.3.4-IA-2	July 14, 2000	August 29, 2000	Instrument Air System
2.3.4-IN-1	July 14, 2000	August 29, 2000	Insulation System
2.3.4-IN-2	July 14, 2000	August 29, 2000	Insulation System
2.3.4-IN-3	July 14, 2000	August 29, 2000	Insulation System
2.3.4-IN-4	July 14, 2000	August 29, 2000	Insulation System
2.3.4-IN-5	July 14, 2000	August 29, 2000	Insulation System
2.3.4-IN-6	July 14, 2000	August 29, 2000	Insulation System
2.3.4-IN-7	July 14, 2000	August 29, 2000	Insulation System
2.3.4-IN-8	July 14, 2000	August 29, 2000	Insulation System
2.3.4-OSHVAC-1	July 14, 2000	August 29, 2000	Outside Structures HVAC
2.3.4-PCCW-1	July 14, 2000	August 29, 2000	Primary Containment Chilled Water System
2.3.4-PCCW-2	July 14, 2000	August 29, 2000	Primary Containment Chilled Water System
2.3.4-PSW-1	July 14, 2000	August 29, 2000	Plant Service Water System
2.3.4-PSW-2	July 14, 2000	August 29, 2000	Plant Service Water System
2.3.4-PSW-3	July 14, 2000	August 29, 2000	Plant Service Water System
2.3.4-PSW-4	July 14, 2000	August 29, 2000	Plant Service Water System

2.3.4-PSW-5	July 14, 2000	August 29, 2000	Plant Service Water System
2.3.4-RBHVAC-1	July 14, 2000	August 29, 2000	Reactor Building HVAC
2.3.4-RBHVAC-2	July 14, 2000	August 29, 2000	Reactor Building HVAC
2.3.4-RBHVAC-3	July 14, 2000	August 29, 2000	Reactor Building HVAC
2.3.4-RW-1	July 14, 2000	August 29, 2000	Radwaste System
2.3.4-TSR-1	July 14, 2000	August 29, 2000	Traveling Water Screens/Trash Racks
2.3.4-TV-1	July 14, 2000	August 29, 2000	Tornado Vents
2.3.5-EHC-1	July 14, 2000	August 29, 2000	Electro-Hydraulic Control System
2.3.5-EHC-2	July 14, 2000	August 29, 2000	Electro-Hydraulic Control System
2.3.5-MC-1	July 14, 2000	August 29, 2000	Main Condenser
2.3.5-SPCS-1	July 14, 2000	August 29, 2000	Steam and Power Conversion Systems
2.4-1	July 14, 2000	August 29, 2000	Structures - General
2.4-2	July 14, 2000	August 29, 2000	Structures - General
2.4-3	July 14, 2000	August 29, 2000	Structures - General
2.4-4	July 14, 2000	August 29, 2000	Structures - General
2.4-CRT-1	July 14, 2000	August 29, 2000	Conduits, Raceways, and Trays
2.4-EDGB-1	July 14, 2000	August 29, 2000	EDG Building
2.4-FS-1	July 14, 2000	August 29, 2000	Fuel Storage
2.4-FS-2	July 14, 2000	August 29, 2000	Fuel Storage
2.4-FS-3	July 14, 2000	August 29, 2000	Fuel Storage
2.4-IS-1	July 14, 2000	August 29, 2000	Intake Structure
2.4-IS-2	July 14, 2000	August 29, 2000	Intake Structure
2.4-IS-3	July 14, 2000	August 29, 2000	Intake Structure

2.4-IS-4	July 14, 2000	August 29, 2000	Intake Structure
2.4-PC-1	July 14, 2000	August 29, 2000	Primary Containment
2.4-PC-2	July 14, 2000	August 29, 2000	Primary Containment
2.4-PS-1	July 14, 2000	August 29, 2000	Piping Specialties
2.4-PS-2	July 14, 2000	August 29, 2000	Piping Specialties
2.4-PS-3	July 14, 2000	August 29, 2000	Piping Specialties
2.4-RB-1	July 14, 2000	August 29, 2000	Reactor Building
2.4-RB-2	July 14, 2000	August 29, 2000	Reactor Building
2.4-RB-3	July 14, 2000	August 29, 2000	Reactor Building
2.4-TB-1	July 14, 2000	August 29, 2000	Turbine Building
2.5-ELEC-1	July 28, 2000	October 10, 2000	Electrical
3.1-1	July 28, 2000	October 10, 2000	AMPs - General
3.1-2	July 28, 2000	October 10, 2000	AMPs - General
3.1-3	July 28, 2000	October 10, 2000	AMPs - General
3.1-4	July 28, 2000	October 10, 2000	AMPs - General
3.1-5	July 28, 2000	October 10, 2000	AMPs - General
3.1-6	July 28, 2000	October 10, 2000	AMPs - General
3.1-7	July 28, 2000	October 10, 2000	AMPs - General
3.1.1-1	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-2	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-3	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-4	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-5	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-6	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control

3.1.1-7	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-8	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-9	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-10	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-11	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.1-12	July 28, 2000	October 10, 2000	Reactor Water Chemistry Control
3.1.2-1	July 28, 2000	October 10, 2000	CCW Chemistry Control
3.1.2-2	July 28, 2000	October 10, 2000	CCW Chemistry Control
3.1.2-3	July 28, 2000	October 10, 2000	CCW Chemistry Control
3.1.2-4	July 28, 2000	October 10, 2000	CCW Chemistry Control
3.1.2-5	July 28, 2000	October 10, 2000	CCW Chemistry Control
3.1.2-6	July 28, 2000	October 10, 2000	CCW Chemistry Control
3.1.2-7	July 28, 2000	October 10, 2000	CCW Chemistry Control
3.1.3-1	July 28, 2000	October 10, 2000	Diesel Fuel Oil Testing
3.1.3-2	July 28, 2000	October 10, 2000	Diesel Fuel Oil Testing
3.1.3-3	July 28, 2000	October 10, 2000	Diesel Fuel Oil Testing
3.1.3-4	July 28, 2000	October 10, 2000	Diesel Fuel Oil Testing
3.1.4-1	July 28, 2000	October 10, 2000	PSW and RHRSW Chemistry Control

3.1.4-2	July 28, 2000	October 10, 2000	PSW and RHRSW Chemistry Control
3.1.4-3	July 28, 2000	October 10, 2000	PSW and RHRSW Chemistry Control
3.1.4-4	July 28, 2000	October 10, 2000	PSW and RHRSW Chemistry Control
3.1.4-5	July 28, 2000	October 10, 2000	PSW and RHRSW Chemistry Control
3.1.4-6	July 28, 2000	October 10, 2000	PSW and RHRSW Chemistry Control
3.1.4-7	July 28, 2000	October 10, 2000	PSW and RHRSW Chemistry Control
3.1.4-8	July 28, 2000	October 10, 2000	PSW and RHRSW Chemistry Control
3.1.5-1	July 28, 2000	October 10, 2000	Fuel Pool Chemistry Control
3.1.5-2	July 28, 2000	October 10, 2000	Fuel Pool Chemistry Control
3.1.5-3	July 28, 2000	October 10, 2000	Fuel Pool Chemistry Control
3.1.5-4	July 28, 2000	October 10, 2000	Fuel Pool Chemistry Control
3.1.5-5	July 28, 2000	October 10, 2000	Fuel Pool Chemistry Control
3.1.5-6	July 28, 2000	October 10, 2000	Fuel Pool Chemistry Control
3.1.6-1	July 28, 2000	October 10, 2000	Demineralized Water and CST Chemistry Control
3.1.7-1	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control
3.1.7-2	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control
3.1.7-3	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control

3.1.7-4	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control
3.1.7-5	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control
3.1.7-6	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control
3.1.7-7	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control
3.1.7-8	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control
3.1.7-9	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control
3.1.7-10	July 28, 2000	October 10, 2000	Suppression Pool Chemistry Control
3.1.8-1	July 28, 2000	October 10, 2000	CAP
3.1.8-2	July 28, 2000	October 10, 2000	CAP
3.1.9-1	July 28, 2000	October 10, 2000	ISI Program
3.1.9-2	July 28, 2000	October 10, 2000	ISI Program
3.1.9-3	July 28, 2000	October 10, 2000	ISI Program
3.1.9-4	July 28, 2000	October 10, 2000	ISI Program
3.1.9-5	July 28, 2000	October 10, 2000	ISI Program
3.1.9-6	July 28, 2000	October 10, 2000	ISI Program
3.1.9-7	July 28, 2000	October 10, 2000	ISI Program
3.1.10-1	July 28, 2000	October 10, 2000	Overhead Crane and Refueling Platform Inspections
3.1.10-2	July 28, 2000	October 10, 2000	Overhead Crane and Refueling Platform Inspections
3.1.10-3	July 28, 2000	October 10, 2000	Overhead Crane and Refueling Platform Inspections
3.1.10-4	July 28, 2000	October 10, 2000	Overhead Crane and Refueling Platform Inspections

3.1.10-5	July 28, 2000	October 10, 2000	Overhead Crane and Refueling Platform Inspections
3.1.10-6	July 28, 2000	October 10, 2000	Overhead Crane and Refueling Platform Inspections
3.1.11-1	July 28, 2000	October 10, 2000	Torque Activities
3.1.11-2	July 28, 2000	October 10, 2000	Torque Activities
3.1.11-3	July 28, 2000	October 10, 2000	Torque Activities
3.1.11-4	July 28, 2000	October 10, 2000	Torque Activities
3.1.11-5	July 28, 2000	October 10, 2000	Torque Activities
3.1.12-1	July 28, 2000	October 10, 2000	CCTLP
3.1.12-2	July 28, 2000	October 10, 2000	CCTLP
3.1.12-3	July 28, 2000	October 10, 2000	CCTLP
3.1.12-4	July 28, 2000	October 10, 2000	CCTLP
3.1.12-5	July 28, 2000	October 10, 2000	CCTLP
3.1.12-6	July 28, 2000	October 10, 2000	CCTLP
3.1.13-1	July 28, 2000	October 10, 2000	PSW and RHRSW Inspection Program
3.1.13-2	July 28, 2000	October 10, 2000	PSW and RHRSW Inspection Program
3.1.13-3	July 28, 2000	October 10, 2000	PSW and RHRSW Inspection Program
3.1.13-4	July 28, 2000	October 10, 2000	PSW and RHRSW Inspection Program
3.1.13-5	July 28, 2000	October 10, 2000	PSW and RHRSW Inspection Program
3.1.14-1	July 28, 2000	October 10, 2000	Primary Containment Leakage Rate Testing Program
3.1.14-2	July 28, 2000	October 10, 2000	Primary Containment Leakage Rate Testing Program
3.1.15-1	July 28, 2000	October 10, 2000	BWRVIP

3.1.15-2	July 28, 2000	October 10, 2000	BWRVIP
3.1.15-3	July 28, 2000	October 10, 2000	BWRVIP
3.1.15-4	July 28, 2000	October 10, 2000	BWRVIP
3.1.15-5	July 28, 2000	October 10, 2000	BWRVIP
3.1.16-1	July 28, 2000	October 10, 2000	Wetted Cable Activities
3.1.16-2	July 28, 2000	October 10, 2000	Wetted Cable Activities
3.1.16-3	July 28, 2000	October 10, 2000	Wetted Cable Activities
3.1.16-4	July 28, 2000	October 10, 2000	Wetted Cable Activities
3.1.16-5	July 28, 2000	October 10, 2000	Wetted Cable Activities
3.1.17-1	July 28, 2000	October 10, 2000	RPV Monitoring Program
3.1.18-1	July 28, 2000	October 10, 2000	Fire Protection Activities
3.1.18-2	July 28, 2000	October 10, 2000	Fire Protection Activities
3.1.18-3	July 28, 2000	October 10, 2000	Fire Protection Activities
3.1.18-4	July 28, 2000	October 10, 2000	Fire Protection Activities
3.1.18-5	July 28, 2000	October 10, 2000	Fire Protection Activities
3.1.18-6	July 28, 2000	October 10, 2000	Fire Protection Activities
3.1.18-7	July 28, 2000	October 10, 2000	Fire Protection Activities
3.1.18-8	July 28, 2000	October 10, 2000	Fire Protection Activities
3.1.18-9	July 28, 2000	October 10, 2000	Fire Protection Activities

3.1.18-10	July 28, 2000	October 10, 2000	Fire Protection Activities
3.1.19-1	July 28, 2000	October 10, 2000	FAC Program
3.1.19-2	July 28, 2000	October 10, 2000	FAC Program
3.1.19-3	July 28, 2000	October 10, 2000	FAC Program
3.1.19-4	July 28, 2000	October 10, 2000	FAC Program
3.1.19-5	July 28, 2000	October 10, 2000	FAC Program
3.1.19-6	July 28, 2000	October 10, 2000	FAC Program
3.1.19-7	July 28, 2000	October 10, 2000	FAC Program
3.1.19-8	July 28, 2000	October 10, 2000	FAC Program
3.1.20-1	July 28, 2000	October 10, 2000	Protective Coatings Program
3.1.20-2	July 28, 2000	October 10, 2000	Protective Coatings Program
3.1.20-3	July 28, 2000	October 10, 2000	Protective Coatings Program
3.1.21-1	July 28, 2000	October 10, 2000	Equipment and Piping Insulation Monitoring Program
3.1.21-2	July 28, 2000	October 10, 2000	Equipment and Piping Insulation Monitoring Program
3.1.21-3	July 28, 2000	October 10, 2000	Equipment and Piping Insulation Monitoring Program
3.1.21-4	July 28, 2000	October 10, 2000	Equipment and Piping Insulation Monitoring Program
3.1.21-5	July 28, 2000	October 10, 2000	Equipment and Piping Insulation Monitoring Program
3.1.22-1	July 28, 2000	October 10, 2000	Structural Monitoring Program
3.1.22-2	July 28, 2000	October 10, 2000	Structural Monitoring Program

3.1.22-3	July 28, 2000	October 10, 2000	Structural Monitoring Program
3.1.22-4	July 28, 2000	October 10, 2000	Structural Monitoring Program
3.1.22-5	July 28, 2000	October 10, 2000	Structural Monitoring Program
3.1.23-1	July 28, 2000	October 10, 2000	Galvanic Susceptibility Inspections
3.1.23-2	July 28, 2000	October 10, 2000	Galvanic Susceptibility Inspections
3.1.23-3	July 28, 2000	October 10, 2000	Galvanic Susceptibility Inspections
3.1.23-4	July 28, 2000	October 10, 2000	Galvanic Susceptibility Inspections
3.1.23-5	July 28, 2000	October 10, 2000	Galvanic Susceptibility Inspections
3.1.24-1	July 28, 2000	October 10, 2000	Treated Water Systems Piping Inspections
3.1.24-2	July 28, 2000	October 10, 2000	Treated Water Systems Piping Inspections
3.1.24-3	July 28, 2000	October 10, 2000	Treated Water Systems Piping Inspections
3.1.24-4	July 28, 2000	October 10, 2000	Treated Water Systems Piping Inspections
3.1.24-5	July 28, 2000	October 10, 2000	Treated Water Systems Piping Inspections
3.1.24-6	July 28, 2000	October 10, 2000	Treated Water Systems Piping Inspections

3.1.25-1	July 28, 2000	October 10, 2000	Gas Systems Components Inspections
3.1.25-2	July 28, 2000	October 10, 2000	Gas Systems Components Inspections
3.1.25-3	July 28, 2000	October 10, 2000	Gas Systems Components Inspections
3.1.25-4	July 28, 2000	October 10, 2000	Gas Systems Components Inspections
3.1.26-1	July 28, 2000	October 10, 2000	CST Inspections
3.1.26-2	July 28, 2000	October 10, 2000	CST Inspections
3.1.26-3	July 28, 2000	October 10, 2000	CST Inspections
3.1.27-1	July 28, 2000	October 10, 2000	Passive Components Inspection Activities
3.1.27-2	July 28, 2000	October 10, 2000	Passive Components Inspection Activities
3.1.28-1	July 28, 2000	October 10, 2000	RHR Heat Exchanger Augmented Inspection and Testing Program
3.1.28-2	July 28, 2000	October 10, 2000	RHR Heat Exchanger Augmented Inspection and Testing Program
3.1.28-3	July 28, 2000	October 10, 2000	RHR Heat Exchanger Augmented Inspection and Testing Program
3.1.28-4	July 28, 2000	October 10, 2000	RHR Heat Exchanger Augmented Inspection and Testing Program
3.1.28-5	July 28, 2000	October 10, 2000	RHR Heat Exchanger Augmented Inspection and Testing Program

3.1.28-6	July 28, 2000	October 10, 2000	RHR Heat Exchanger Augmented Inspection and Testing Program
3.1.28-7	July 28, 2000	October 10, 2000	RHR Heat Exchanger Augmented Inspection and Testing Program
3.1.29-1	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-2	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-3	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-4	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-5	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-6	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-7	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-8	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-9	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-10	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program

3.1.29-11	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-12	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.1.29-13	July 28, 2000	October 10, 2000	Torus Submerged Components Inspection Program
3.2.3.1-1	July 28, 2000	October 10, 2000	RCS
3.2.3.1-2	July 28, 2000	October 10, 2000	RCS
3.2.3.2-1	July 28, 2000	October 10, 2000	RCS
3.2.3.2-2	July 28, 2000	October 10, 2000	RCS
3.2.3.2-3	July 28, 2000	October 10, 2000	RCS
3.2.3.2-4	July 28, 2000	October 10, 2000	RCS
3.2.3.2-5	July 28, 2000	October 10, 2000	RCS
3.2.3.2-6	July 28, 2000	October 10, 2000	RCS
3.2.3.2-7	July 28, 2000	October 10, 2000	RCS
3.2.3.2-8	July 28, 2000	October 10, 2000	RCS
3.3-CS-1	July 28, 2000	October 10, 2000	Core Spray
3.3-HPCI-1	July 28, 2000	October 10, 2000	HPCI
3.3-HPCI-2	July 28, 2000	October 10, 2000	HPCI
3.3-HPCI-3	July 28, 2000	October 10, 2000	HPCI
3.3-HPCI-4	July 28, 2000	October 10, 2000	HPCI
3.3-HPCI-5	July 28, 2000	October 10, 2000	HPCI
3.3-HPCI-6	July 28, 2000	October 10, 2000	HPCI
3.3-HPCI-7	July 28, 2000	October 10, 2000	HPCI
3.3-HPCI-8	July 28, 2000	October 10, 2000	HPCI
3.3-HPCI-9	July 28, 2000	October 10, 2000	HPCI
3.3-HR-1	July 28, 2000	October 10, 2000	Post-LOCA Hydrogen Recombiner System

3.3-P&I-1	July 28, 2000	October 10, 2000	Primary Containment Purge & Inerting System
3.3-P&I-2	July 28, 2000	October 10, 2000	Primary Containment Purge & Inerting System
3.3-RCIC-1	July 28, 2000	October 10, 2000	RCIC System
3.3-RCIC-2	July 28, 2000	October 10, 2000	RCIC System
3.3-RCIC-3	July 28, 2000	October 10, 2000	RCIC System
3.3-RCIC-4	July 28, 2000	October 10, 2000	RCIC System
3.3-RCIC-5	July 28, 2000	October 10, 2000	RCIC System
3.3-RCIC-6	July 28, 2000	October 10, 2000	RCIC System
3.3-RCIC-7	July 28, 2000	October 10, 2000	RCIC System
3.3-SGTS-1	July 28, 2000	October 10, 2000	Standby Gas Treatment System
3.3-SGTS-2	July 28, 2000	October 10, 2000	Standby Gas Treatment System
3.4-1	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-2	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-3	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-4	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-5	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-6	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-7	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-8	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-9	July 28, 2000	October 10, 2000	Auxiliary Systems - General

3.4-10	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-11	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-12	July 28, 2000	October 10, 2000	Auxiliary Systems - General
3.4-CBHVAC-1	July 28, 2000	October 10, 2000	Control Building HVAC
3.4-CHE-1	July 28, 2000	October 10, 2000	Cranes, Hoists, and Elevators
3.4-COND-1	July 28, 2000	October 10, 2000	Condensate Transfer and Storage System
3.4-CRD-1	July 28, 2000	October 10, 2000	Control Rod Drive System
3.4-CRD-2	July 28, 2000	October 10, 2000	Control Rod Drive System
3.4-DPS-1	July 28, 2000	October 10, 2000	Drywell Pneumatic Systems
3.4-FPS-1	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-2	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-3	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-4	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-5	July 28, 2000	October 10, 2000	
3.4-FPS-6	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-7	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-8	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-9	July 28, 2000	October 10, 2000	Fire Protection System

3.4-FPS-10	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-11	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-12	July 28, 2000	October 10, 2000	Fire Protection System
3.4-FPS-13	July 28, 2000	October 10, 2000	Fire Protection System
3.4-IA-1	July 28, 2000	October 10, 2000	Instrument Air System
3.4-IN-1	July 28, 2000	October 10, 2000	Insulation System
3.4-PSW-1	July 28, 2000	October 10, 2000	Plant Service Water System
3.4-PSW-2	July 28, 2000	October 10, 2000	Plant Service Water System
3.4-PSW-3	July 28, 2000	October 10, 2000	Plant Service Water System
3.4-PSW-4	July 28, 2000	October 10, 2000	Plant Service Water System
3.4-PSW-5	July 28, 2000	October 10, 2000	Plant Service Water System
3.4-PSW-6	July 28, 2000	October 10, 2000	Plant Service Water System
3.4-RBHVAC-1	July 28, 2000	October 10, 2000	Reactor Building HVAC
3.4-RE-1	July 28, 2000	October 10, 2000	Refueling Equipment
3.4-RE-2	July 28, 2000	October 10, 2000	Refueling Equipment
3.4-RE-3	July 28, 2000	October 10, 2000	Refueling Equipment
3.4-SS-1	July 28, 2000	October 10, 2000	Sampling System
3.4-TSR-1	July 28, 2000	October 10, 2000	Traveling Water Screens/Trash Racks System
3.4-TSR-2	July 28, 2000	October 10, 2000	Traveling Water Screens/Trash Racks System

3.5-EHC-1	July 28, 2000	October 10, 2000	Electro-Hydraulic Control System
3.5-MC-1	July 28, 2000	October 10, 2000	Main Condenser
3.5-MC-2	July 28, 2000	October 10, 2000	Main Condenser
3.6-1	July 28, 2000	October 10, 2000	Structures
3.6-2	July 28, 2000	October 10, 2000	Structures
3.6-3	July 28, 2000	October 10, 2000	Structures
3.6-4	July 28, 2000	October 10, 2000	Structures
3.6-5	July 28, 2000	October 10, 2000	Structures
3.6-6	July 28, 2000	October 10, 2000	Structures
3.6-7	July 28, 2000	October 10, 2000	Structures
3.6-8	July 28, 2000	October 10, 2000	Structures
3.6-9	July 28, 2000	October 10, 2000	Structures
3.6-10	July 28, 2000	October 10, 2000	Structures
3.6-11	July 28, 2000	October 10, 2000	Structures
3.6-12	July 28, 2000	October 10, 2000	Structures
3.6-13	July 28, 2000	October 10, 2000	Structures
3.6-14	July 28, 2000	October 10, 2000	Structures
3.6-15	July 28, 2000	October 10, 2000	Structures
3.6-16	July 28, 2000	October 10, 2000	Structures
3.6-17	July 28, 2000	October 10, 2000	Structures
3.6-18	July 28, 2000	October 10, 2000	Structures
3.6-19	July 28, 2000	October 10, 2000	Structures
3.6-20	July 28, 2000	October 10, 2000	Structures
3.6-21	July 28, 2000	October 10, 2000	Structures
3.6-22	July 28, 2000	October 10, 2000	Structures
3.6-23	July 28, 2000	October 10, 2000	Structures
3.6-24	July 28, 2000	October 10, 2000	Structures
3.6-25	July 28, 2000	October 10, 2000	Structures

3.6-26	July 28, 2000	October 10, 2000	Structures
3.6-27	July 28, 2000	October 10, 2000	Structures
3.6-28	July 28, 2000	October 10, 2000	Structures
3.6-29	July 28, 2000	October 10, 2000	Structures
3.6-30	July 28, 2000	October 10, 2000	Structures
3.6-31	July 28, 2000	October 10, 2000	Structures
3.6-32	July 28, 2000	October 10, 2000	Structures
3.6-33	July 28, 2000	October 10, 2000	Structures
3.6-34	July 28, 2000	October 10, 2000	Structures
3.6-35	July 28, 2000	October 10, 2000	Structures
3.6-36	July 28, 2000	October 10, 2000	Structures
3.6-37	July 28, 2000	October 10, 2000	Structures
3.6-38	July 28, 2000	October 10, 2000	Structures
3.6-39	July 28, 2000	October 10, 2000	Structures
3.6-40	July 28, 2000	October 10, 2000	Structures
3.6-41	July 28, 2000	October 10, 2000	Structures
3.6-42	July 28, 2000	October 10, 2000	Structures
3.6-43	July 28, 2000	October 10, 2000	Structures
3.6-44	July 28, 2000	October 10, 2000	Structures
3.6-45	July 28, 2000	October 10, 2000	Structures
3.6-46	July 28, 2000	October 10, 2000	Structures
3.6-47	July 28, 2000	October 10, 2000	Structures
3.6-48	July 28, 2000	October 10, 2000	Structures
3.6-49	July 28, 2000	October 10, 2000	Structures
3.6-50	July 28, 2000	October 10, 2000	Structures
3.6-51	July 28, 2000	October 10, 2000	Structures
3.6-52	July 28, 2000	October 10, 2000	Structures
3.6-53	July 28, 2000	October 10, 2000	Structures

3.6-54	July 28, 2000	October 10, 2000	Structures
3.6-55	July 28, 2000	October 10, 2000	Structures
4.1-1	July 28, 2000	October 10, 2000	TLAA
4.1-2	July 28, 2000	October 10, 2000	TLAA
4.2-1	July 28, 2000	October 10, 2000	TLAA
4.2-2	July 28, 2000	October 10, 2000	TLAA
4.2-3	July 28, 2000	October 10, 2000	TLAA
4.2-4	July 28, 2000	October 10, 2000	TLAA
4.4-1	July 28, 2000	October 10, 2000	TLAA
4.4-2	July 28, 2000	October 10, 2000	TLAA
4.5-1	July 28, 2000	October 10, 2000	TLAA
4.5-2	July 28, 2000	October 10, 2000	TLAA
4.5-3	July 28, 2000	October 10, 2000	TLAA
4.5-4	July 28, 2000	October 10, 2000	TLAA
4.6-1	July 28, 2000	October 10, 2000	TLAA
4.6-2	July 28, 2000	October 10, 2000	TLAA
4.6-3	July 28, 2000	October 10, 2000	TLAA
4.6-4	July 28, 2000	October 10, 2000	TLAA
4.7-1	July 28, 2000	October 10, 2000	TLAA

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Docket Nos. 50-321 and 50-366

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Docket Numbers 50-321 and 50-366

11. ABSTRACT *(200 words or less)*

This document is a safety evaluation report regarding the application to renew the operating licenses for the Edwin I. Hatch Nuclear Plants, Units 1 and 2 (Plant Hatch), which was filed by the Southern Nuclear Operating Company, Inc. by letter dated February 29, 2000. The Office of Nuclear Reactor Regulation has reviewed the Plant Hatch license renewal application for compliance with the requirements of Title 10 of the Code of Federal Regulations, Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and prepared this report to document its findings.

In its submittal of February 29, 2000, Southern Nuclear Operating Company requested renewal of the operating licenses (License Nos. DPR-57 and NPF-5) for Units 1 and 2, respectively. These licenses were issued under Sections 104 and 103 respectively, of the Atomic Energy Act of 1954, as amended, for a period of 20 years beyond the current license expiration dates of August 6, 2014 for Unit 1, and June 13, 2018 for Unit 2. Plant Hatch is located in Appling County, Georgia, and consists of two General Electric (GE) boiling-water reactor (BWR) nuclear steam supply systems designed to generate 2763 MW-thermal, or approximately 900 MW-electric.

The NRC Plant Hatch license renewal project manager is William F. Burton. Mr. Burton may be contacted by calling (301) 415-2853 or by writing to the License Renewal and Standardization Branch, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

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