NRC FORM 464 Part I
(6-1998)

U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) / PRIVACY ACT (PA) REQUEST

FOIA/PA	RESPONSE NUMBER
2000-0191	1

RESPONSE TYPE: ✓ FINAL ☐ PARTIAL

REQUESTER: Mr. Andrew Wagner

DATE: JUN 05 2000

PART I. -- INFORMATION RELEASED

☐ No additional agency records subject to the request have been located.

☐ Requested records are available through another public distribution program. See Comments section.

✓ APPENDICES A — Agency records subject to the request that are identified in the listed appendices are already available for public inspection and copying at the NRC Public Document Room.

✓ APPENDICES B — Agency records subject to the request that are identified in the listed appendices are being made available for public inspection and copying at the NRC Public Document Room.

✓ Enclosed is information on how you may obtain access to and the charges for copying records located at the NRC Public Document Room, 2120 L Street, NW, Washington, DC.

✓ APPENDICES B — Agency records subject to the request are enclosed.

☐ Records subject to the request that contain information originated by or of interest to another Federal agency have been referred to that agency (see comments section) for a disclosure determination and direct response to you.

☐ We are continuing to process your request.

☐ See Comments.

PART I.A -- FEES

AMOUNT * $

* See comments for details

☐ You will be billed by NRC for the amount listed.

☐ You will receive a refund for the amount listed.

☐ None. Minimum fee threshold not met.

✓ Fees waived.

PART I.B -- INFORMATION NOT LOCATED OR WITHHELD FROM DISCLOSURE

☐ No agency records subject to the request have been located.

☐ Certain information in the requested records is being withheld from disclosure pursuant to the exemptions described in and for the reasons stated in Part II.

☐ This determination may be appealed within 30 days by writing to the FOIA/PA Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Clearly state on the envelope and in the letter that it is a "FOIA/PA Appeal."

PART I.C COMMENTS (Use attached Comments continuation page if required)

In a telephone conversation on May 3, 2000, with Nina Pugh , you rescoped your request to records relating to generic renewal issues developed from industry comments on the 1995 rule change and subsequent guidance. These records are identified on Appendices A and B. Copies of Appendix B records are enclosed. For your information, the NRC does not provide copies of or waive fees for records that have already been made publicly available at NRC's Public Document Room, (PDR). Copies of Appendix A records one through six may be obtained by contacting the PDR directly. The records identified on Appendix A with a ML accession number are publicly available in the NRC's Public Electronic Reading Room at http://www.nrc.gov/NRC/ADAMS/index.html. If you need assistance in obtaining these records, please contact the PDR at (202)634-3273, or 1-800-397-4209, or by e-mail to pdr@nrc.gov.

This completes NRC's action on your FOIA.

SIGNATURE - FREEDOM OF INFORMATION ACT AND PRIVACY ACT OFFICER

Carol Ann Reed *Carol Ann Reed*

FOIA No. 2000-0191

APPENDIX A
RECORDS ALREADY AVAILABLE IN THE PDR

NO.	DATE	ACCESSION NUMBER	DESCRIPTION/(PAGE COUNT)
1.	10/27/97	9700040157	Letter to NRC from H.L. Sumner, Southern Nuclear Operating, Co., Inc. Re: Request for waiver of review fees (3 pages)
2.	1/15/98	9801230066	Letter to NRC from H. L. Sumner, Southern Nuclear Operating, Co., Inc. Re: Product submittals to NRC (3 pages)
3.	04/13/98	9804220149	Letter to NRC from H. L. Sumner, Southern Nuclear Operating, Co., Inc. Re: E.I. Hatch process for implementing technical requirements (3 pages)
4.	05/03/98	9805060036	Letter to H. L. Sumner, Southern Nuclear Operating, Co., Inc. from S. Collins, NRC Re: License renewal application for plant (2 pages)
5.	01/07/99	9901130111	Letter to NRC from H. L. Sumner, Southern Nuclear Operating, Co., Inc. Re: EHNP Intake Structure License Report (2 pages)
6.	01/25/99	9903160142	Letter to S. Jackson, NRC from W.G. Hairston, Southern Nuclear Operating, Co., Inc. Re: Utilities support to NRCs recent initiatives to streamline the hearing process (3 pages)

7.	11/12/99	ML993270222	Ltr to NRC from H.L. Sumner, Southern Nuclear Operating Co. Re: Edwin Hatch Nuclear Plant request for exemption (24 pages)
8.	1/24/00	ML003677239	Ltr to H.L. Sumner, Southern Nuclear Operating Co., from B. Shelton, NRC Re: Edwin Hatch Nuclear Plant request for exemption (5 pages)
9.	02/29/00	ML003688151	Ltr to NRC from H.L. Sumner, Southern Nuclear Operating Co., Re: Edwin Hatch Nuclear Plant Application for Renewed Operating Licenses (1122 pages)
10.	02/29/00	ML003688222	Ltr to NRC from H.L. Sumner, Southern Nuclear Operating Co., Re: Edwin Hatch Nuclear Plant (24 pages)
11.	03/03/00	ML003688811	Ltr to H.L. Sumner, Southern Nuclear Operating Co. from C. Grimes, NRC re: Receipt of the Edwin Hatch Unit Nos. 1 & 2, License renewal application & assignment of a project manager (4 pages)
12.	03/24/00	ML003695605	Ltr to H.L. Sumner, Southern Nuclear Operating Co. from D. Matthews, NRC RE: Determination of acceptability and sufficiency for docketing Edwin Hatch Nuclear Plant (12 pages)

FOIA No. 2000-0191

APPENDIX B
RECORDS BEING RELEASED IN THEIR ENTIRETY
(If copyrighted identify with *)

NO.	DATE	DESCRIPTION/(PAGE COUNT)
1.	01/23/98	Handout re: Presentation on Hatch License Renewal Program BWROG License Renewal Program by Southern Nuclear License Renewal Manager (6 pages)
2.	02/01/98	Handout regarding Sixth International Conference on Nuclear Engineering Aging Assessment, Life Extension and License Renewal (8 pages)
3.	04/13/98	Letter to Chairman Jackson, NRC from H.L. Sumner, Jr., Re: Baltimore Gas & Electric Company license Renewal Application for Calvert Cliffs Nuclear Power Plant (2 pages)
4.	08/18/98	Handout from NRC/SNC Meeting Re" Plant Hatch License Renewal Process (22 pages)
5.	04/13/99	Handout from Southern Nuclear Hatch License Renewal Project Manager (11 pages)
6.	05/14/99	Letter to J. Hoyle, NRC from H. Sumner, Southern Nuclear Operating Co Re: Edwin Hatch Nuclear Plant Recirculation System Pressure Boundary Licensing Report (56 pages)

7.	02/03/2000	Memo to C. Grimes, NRC from R. Anand, NRC re: Meeting with Southern Nuclear Operating Co (SNC) on license renewal for Hatch, Units 1 and 2 (7 pages)
8.	02/22/2000	Memo to Southern Nuclear Operating Co, Inc. from W. Burton, NRC., Re: Handouts and Summary of meeting with Southern Nuclear Operating Co., Inc. (78 pages)
9.	03/01/2000	E-mail to J. Davis, Southern Nuclear Operating Co., Inc., from J. Wilson, NRC; Re: License Renewal Application (1 page)
10.	03/01/2000	E-mail to J. Wilson, NRC from J. Davis, Southern Nuclear Operating Co., Inc., Re: State and Federal Agency Contacts (2 pages)
11.	03/06/2000	E-mail to J. Wilson, NRC from J. Davis, Southern Nuclear Operating Co., Inc., Re: Local Contacts (4 pages)
12.	03/13/2000	E-mail to J. Wilson, NRC from J. Davis, Southern Nuclear Operating Co., Inc., Re: Environmental Report Part 1-4 (4 page)
13.	03/24/2000	E-mail to W. Burton, NRC from R. Baker, Southern Nuclear Operating Co., Inc., Re: Matrix of Commodities to Programs/Activities (6 pages)
14.	04/04/2000	Letter to L. Sumner, Southern Nuclear Operating Co., from D. Matthews, NRC; Re: Notice of Intent (9 pages)

15.	04/14/2000	E-mail to J. Davis Southern Nuclear Operating Co., Inc., from J. Wilson, NRC; Re: Questions on SAMA (1 page)
16.	05/16/2000	Printout of Generic License Renewal Issues (12 pages)

1/23/98
ACRS Briefing

Presentation

Hatch License Renewal Program
BWROG License Renewal Program

by Charles Pierce
Southern Nuclear License Renewal Manager
BWROG License Renewal Chairman

SOUTHERN COMPANY

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Program Overview

- Organization created to manage license renewal efforts for Plants Hatch, Farley, Vogtle.
- Plant Hatch is first focus due to earliest operating license expiration date (Unit 1 - 2014).
- Fundamental reasons for beginning now:
  - ✓ Exercise license renewal to achieve an efficient, stable, and predictable process.
  - ✓ Issuing renewed license is ten year process.
  - ✓ Benefits our company's health in light of current issues (e.g. global warming).

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Teaming Initiative

- Involves Southern Nuclear's Hatch, PECO Energy's Peach Bottom, and Northern States Power's Monticello plants.

- Provides for joint use of resources in developing Aging Management Products.

- Process being considered
  - ✓ Hatch to develop AMR for intake structure; Peach Bottom, recirculation piping; Monticello, standby liquid control using consensus process.
  - ✓ Peach Bottom and Monticello then revises their AMR for Hatch Plant
  - ✓ Hatch uses products to develop license submittals on systems.
  - ✓ Peach Bottom and Monticello followup with system submittals once issues are addressed on Hatch docket.

- Reasons for pursuing:
  - ✓ Helps to assure process development that can be applied to other BWRs.
  - ✓ Teaming submittals will indicate efficiency of later reviews
  - ✓ Helps educate other BWRs on existing process.

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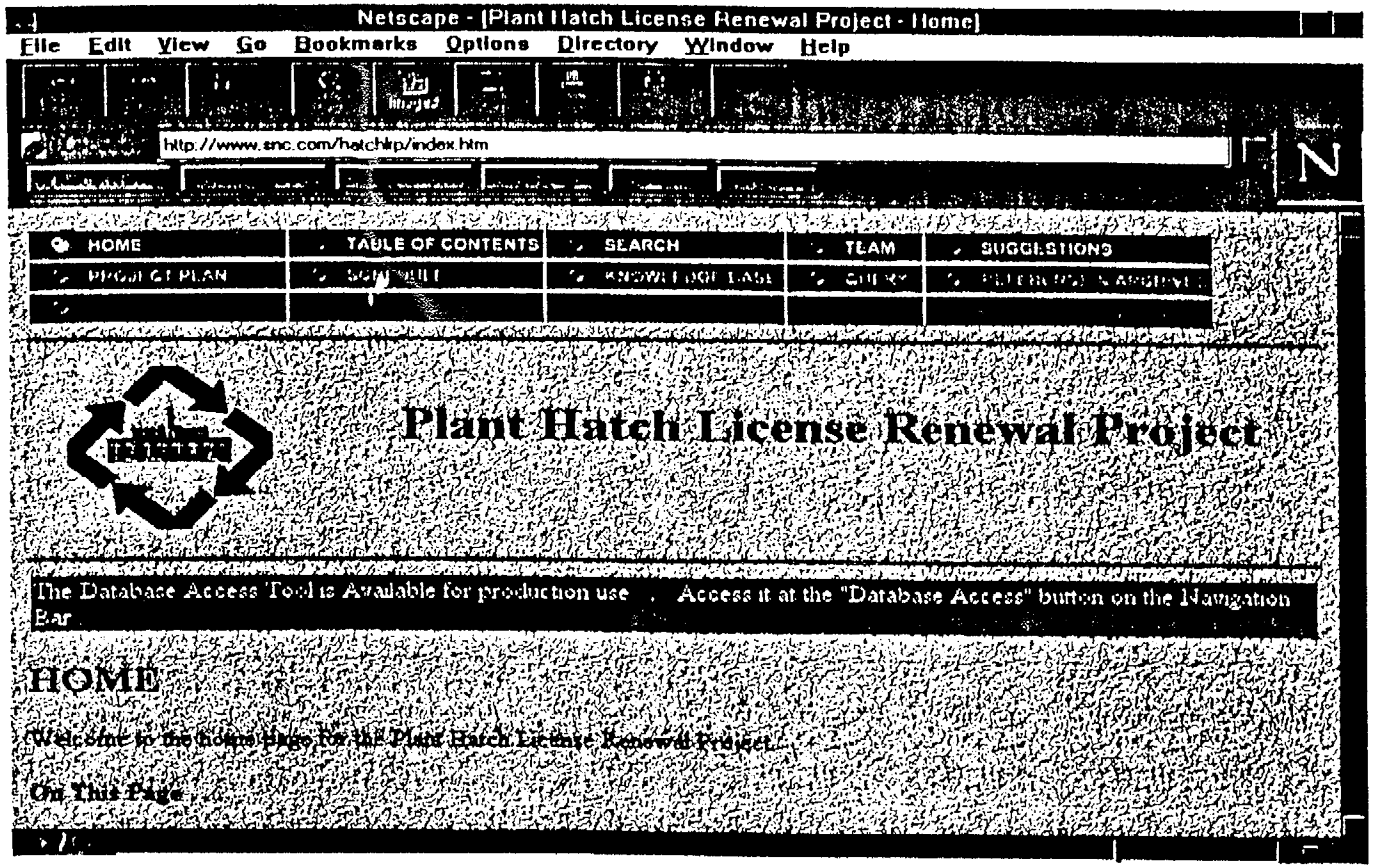
Work Processes - Intranet



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Current Issues and 1998 Deliverables to NRC

- Examples of Issues Raised Through Hatch Activities to Date (Scoping and Screening).
  - ✓ Complex assemblies
  - ✓ Consumables
  - ✓ Special scoping issues (e.g. Seismic II/I, etc.)

- NRC 1998 Currently Planned Deliverables.
  - ✓ First Quarter - Methodology Document
  - ✓ Third Quarter - SLC, Recirc. Piping, Intake Structure Reports
  - ✓ Fourth Quarter - Reactor Vessel Internals Report (depending on BWRVIP status)

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BWROG Activities

- Other Current Activities
  - ✓ Provide partial funding for EPRI investigations (e.g. fatigue, etc.).
  - ✓ Support NEI initiatives (e.g. review of draft SRP, etc.).
  - ✓ Provide partial funding BWR plant activities to pursue renewal (e.g. teaming).
  - ✓ Developing strategy to use materials in plants being decommissioned.

~1/98

ICONE-6
Sixth International Conference on Nuclear Engineering
Aging Assessment, Life Extension and License Renewal

THE PLANT HATCH LICENSE RENEWAL PROGRAM

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ABSTRACT

A revised license renewal rule was issued in May 1995. It addressed many of the industry's concerns about the stability and practicality of the original rule. The industry's attention since that time has been focused on developing generic industry guidance for implementing the rule and testing the guidance through various demonstration programs and work products in conjunction with the NRC. These activities show that implementation issues continue to exist. Some of the more significant issues deal with the process to demonstrate the effectiveness of existing plant programs and the level of detail that will be required in the license renewal application.

Since the issuance of the rule, the NRC has issued a draft license renewal regulatory guide (RG) and standard review plan (SRP). These documents clarify the NRC's positions on several of the implementation issues and identify several additional issues. Further NRC and industry

B/2

interaction will be needed to successfully resolve these issues. The NRC plans to maintain the RG and SRP in draft form until completion of several plant applications' reviews.

Southern Nuclear Operating Company (SNC) has begun development work on a license renewal application for Plant Hatch Units 1 and 2. Plant Hatch Units 1 and 2 are BWR 4, Mark I plants whose operating licenses expire in 2014 and 2018 respectively. Many factors influenced SNC's decision to move ahead with the application development for Plant Hatch at this time. One of these factors was to assure that SNC could continue to play a direct role in contributing to the successful resolution of the RG and SRP issues previously identified. Other factors deal with the length of time required to obtain a renewed license (estimated at about ten years) and the overall financial competitiveness surrounding nuclear power. The "competition issue" is focused on two principal areas. The first area deals with the impact of major capital expenses on a decision to shut down or continue operation toward the end of a nuclear plant's life. The second area deals with the uncertainty of the overall impact of fossil emissions on the cost of fossil generation, and the need to maintain options consistent with changing national policy. SNC will decide on whether to actually submit the application to the NRC when the application is completed.

The Plant Hatch initiative also involves teaming with other BWRs to develop the license renewal technology within the BWR fleet, and to support Plant Hatch by providing an oversight role of application progress. This initiative also involves early interaction with the NRC to address areas of uncertainty in the process. This early interaction is expected to focus on methodology and technical submittals. One of the important areas to be addressed early in the process deals with establishing adequacy of existing programs. Two important programs under consideration for early submittal are the Maintenance Rule structural monitoring program and various Boiling Water Reactor Vessel and Internals Project (BWRVIP) reports.

The BWRVIP was established to address current operating term issues for the BWR reactor vessel and internals; however the BWRVIP recognized that an effective use of technical and licensing resources was to also address the extended operating term. The BWRVIP has prepared an appendix to the inspection and flaw evaluation guidance for each of the vessel internal components that demonstrates how the guidance also addresses the requirements of the license renewal rule. Plant Hatch expects that conducting the NRC reviews of these BWRVIP guidelines for both current and renewal terms at the same time will help to better define the relationships between the current and renewal terms for existing programs under regulatory oversight.

I. INTRODUCTION

Several key developments have occurred with respect to evaluating the benefit of proceeding with the license renewal option. Issuing the Rule and regulatory and industry implementation guidance have been positive developments. National policy decisions may contribute to an improved competitive position for nuclear power. However, there are some significant areas of uncertainty that may affect the reliability of the estimated cost to obtain the renewed license and to implement the licensing commitments during the current and extended operating terms. NRC resource allocation decisions have adversely affected the opportunity for timely generic resolution of the associated process and technical issues through owners' group initiatives and submittals. For the most part, the NRC is focusing its resources on the review of plant-specific submittals.

These developments, and other factors, have lead SNC to decide to proceed with its license renewal program at Plant Hatch. The program strategies, and participation in teaming relationships, are expected to provide opportunities for early interaction with the NRC. The goal is to resolve, or at least better define, the areas of uncertainty and to assess their impact on the Plant Hatch program.

II. THE RULE AND ISSUES AFFECTING IMPLEMENTATION

In December 1991, the Nuclear Regulatory Commission (NRC) published 10 CFR Part 54 to establish the requirements governing nuclear plant license renewal. In this time period, the industry's initiatives included the license renewal lead plant program, preparation of generic industry reports, addressing the environmental requirements, and other activities sponsored by the Nuclear Energy Institute (NEI), Electric Power Research Institute (EPRI), U.S. Department of Energy (DOE), and the various owners' groups. These actions generally determined that some of the requirements of the rule were not clearly defined and would result in an uncertain regulatory process. The NRC interacted with the industry on these concerns and proceeded to amend the rule. The amended or final license renewal rule was issued in May 1995. It addressed many of the industry's concerns about the practicality of the original rule.

The final rule requires a review of certain plant systems, structures, and components (SSCs) to ultimately determine if the effects of aging are adequately managed for certain structures and components in the period of extended operation. An initial scoping process requires a review of all SSCs to identify those that are within the scope of the rule and their intended functions. An integrated plant assessment process (IPA) is then used to identify those structures and components that require an aging management review (AMR), identify time limited aging analysis (TLAA) issues, perform the AMRs to demonstrate that the affects of aging will be properly managed, and to evaluate the TLAA issues. The purpose of the IPA is to provide reasonable assurance that the structures and components will perform their intended functions consistent with the CLB throughout the extended operating period. Very little is said in the rule about acceptable scoping and IPA methods. The rule requires that the IPA be documented in a license renewal application and submitted to the NRC for review.

Once the final rule was issued, the industry focused their efforts on developing generic industry guidance, NEI 95-10 Revision 0, for implementing the rule. The Owners' Groups developed various generic products in an effort to further understand and define the IPA process and methods for evaluating time-limited aging analyses. In parallel, BG&E and Duke initiated efforts to develop license renewal applications for Calvert Cliffs and Oconee, respectively. Several plants and the NRC participated in an NEI sponsored demonstration of the generic industry guidance. The demonstration program and the process of NRC and industry interaction identified that the generic guidance was workable. However, it also identified that there are several implementation issues that need to be resolved to ensure a stable licensing process and that the associated plant licensing commitments can be implemented in a cost effective manner.

Some of the more significant issues are the need to provide detailed demonstrations of the effectiveness of "all" existing programs, the acceptability of performance and condition monitoring programs for aging management review, and the overall level of detail that will be required in the license renewal application.

The NRC subsequently issued a draft regulatory guide (RG) and standard review plan (SRP) that, for the most part, adopts the industry guidance, clarifies their position on several of the implementation issues, and exposes several additional issues. Further NRC and industry interaction will be needed to successfully resolve these issues and the issues identified during the demonstration program. The NRC plans to maintain the RG and SRP in a draft form until it has completed the review of several plant unique applications. These documents will then be completed using the lessons learned from these reviews. This would be a reasonable approach, but unfortunately the NRC License Renewal Directorate announced that they would focus there resources on the review of plant-unique applications. Very little, if any, resources would be applied to the review of Owners' Group or other generic products/documents. Therefore, there is a concern that industry-wide participation and considerations in the final resolution of the implementation issues may be very limited.

III. THE STRATEGY AND PROGRAM FOR HATCH

SNC has been involved in license renewal throughout the rule making process and during the development of the generic implementation guidance. Plant Hatch was one of the sites that participated in the NEI sponsored demonstration of the generic industry guidance. SNC strategic evaluations have periodically assessed the costs and benefits of extended operating scenarios for Plant Hatch. These evaluations have shown that the overall financial competitiveness surrounding nuclear power is focused on two principal areas.

The first area deals with the impact of major capital expenses on a decision to shut down or continue operation toward the end of a nuclear plant's life. At any time, a major capital expense could become necessary to continue operation. Late in plant life, such an expense could result in a decision to shutdown rather than incur the expense. License renewal extends the time available to recover such an expense and thereby improves the probability that such an expense will not force the plant to shutdown.

The second area deals with the uncertainty of the overall impact of fossil emissions on the cost of fossil generation, and the need to maintain options consistent with changing national policy. The recent treaty in Kyoto, if ratified, requires the U.S. to reduce greenhouse emissions to 7% below 1990 levels between 2008 and 2012. The Department of Energy (DOE) estimates a $50/ton tax will be needed to establish 1990 CO_2 levels, which generally translates into an average 1.00 cent/KWH increased cost for fossil plants. Obviously, added costs would be even higher to get emissions to 7% below 1990 levels. Since nuclear power plants produce no greenhouse emissions, renewal of existing nuclear plants could be a vitally important option for maintaining generation capacities and meeting national emission standards of the future.

Plant Hatch Units 1 and 2 are BWR 4, Mark I plants whose operating licenses expire in 2014 and

2018 respectively. SNC has decided to move ahead with the preparation of a license renewal application for Plant Hatch. The main reason is to assure that SNC can continue to play a direct role in contributing to the successful resolution of the RG and SRP issues previously identified. Because the NRC is currently giving priority to plant specific documents over owner's group submittals, this is the best means SNC has to influence the outcome. Other factors that influenced SNC's decision included the above competitive issues and the length of time required to obtain a renewed license, which is estimated at about ten years. In addition, nuclear power needs assurance there will be a nuclear power future in order to attract talented workforce replacements. Finally, the license renewal process needs to be exercised to achieve stability and predictability.

The Plant Hatch license renewal program is a multi-year program with a goal of having a renewal application ready for submittal by the end of 1999. During the preparation of the application, SNC plans to solicit the NRC's review of several documents addressing key process or technical issues. SNC will decide on whether to submit the application once it is completed.

Activities leading to the development of the plant-specific process and procedural documents are nearly complete. Scoping and screening activities are also proceeding. Completing the IPA and TLAA evaluations for the major plant structures and components is planned for 1998.

IV. TEAMING INITIATIVES

Resolution of the significant implementation issues may affect all plants. They will certainly affect those plants that elect to preserve the license renewal option. Their confidence that license renewal can be a cost-effective option may be eroded if these issues are not favorably resolved. There is also a concern that the resolution of some of these issues could affect current term operating practices, whether or not the plant is pursuing an extended operating license. Therefore it is important that the NRC be presented with a variety of viable implementation approaches. This will ensure that the issue resolutions do not foreclose any practical approaches that satisfy the requirements of the rule and streamline the process based on industry experience or plant-specific considerations. SNC believes there is a relatively small window of opportunity for influencing the NRC's final resolution of the implementation issues. Therefore, a short-term approach of preparing technical reports that address selected implementation topics/issues is needed to allow the NRC to review and understand the basis and merits of alternate approaches/positions.

To address this situation, SNC has initiated a teaming relationship with several other utilities with BWR plants. The following areas of common ground, shared points of view, and/or issues of concern provided the reasons for these utilities to participate in a license renewal-teaming program.

- Each of the utilities have BWR plants.
- There is some level of safety/benefit of more than one BWR proceeding concurrently with license renewal; especially during the licensing review process.
- Teaming provides each utility a measure of assurance that their programs, procedures,

and/or positions are on the right track.

- The utilities have a common philosophy on the important implementation issues.
- The window of opportunity to influence the implementation process is rapidly closing.
- Teaming should provide an opportunity to reduce costs.
- Teaming provides opportunities for meaningful mentoring support.
- A plant specific versus a generic approach/submittal is required to obtain NRC resources.
- Having a history of good performance (plant/regulatory) is essential for license renewal to be a cost-effective option.
- Nuclear is part of the long-term strategy for each of the utilities.

It is recognized that establishing a consensus between the team members on the implementation "details" will require patience and determination. However, it is also recognized by each of the utilities that there is significant benefit in early and broader interaction with the NRC to address the areas of uncertainty. In this context, it was agreed that the goals of the teaming effort should include the following:

- Work to favorably influence the NRC's positions and decision making process with respect to implementing the license renewal rule.
- Work to reduce the costs to prepare the license renewal application and for the NRC's review.
- Ensure that the license renewal applications of the participating utilities could be submitted ahead of the "pack."
- Establish a success path (standard) for other BWRs to follow.
- Apply strategies that reduce the potential for public intervention.
- Promote wider industry involvement.

A full commitment by each of the utilities to proceed in parallel to prepare and submit license renewal applications would be the strongest approach. However, a two phased approach was determined to be the most viable approach. The first phase would be to select structure and component topics that involve important implementation issues and then to work in a collaborative manner to submit application material for each plant on these topics. The second phase would be to prepare and submit a full application. Each utility could do this on their own schedule. The fact that three utilities would commit to submit material on their own docket was believed to be a strong selling point with the NRC.

Other than recognizing plant unique features and commitments, each utility agreed to produce products that are essentially the same. This would include the processes at the various steps of the scoping, screening and IPA results, programs credited in the aging management reviews, TLAA issues and evaluation approach, and treatment of industry issues. The reactor recirculation piping, pumps and valves; standby liquid control system; and intake structure were targeted for the accelerated aging management reviews and the development of "application equivalent" reports that can be submitted for review by the NRC. It is intended that these reports provide pattern documents that can be "directly" used by other team members and BWRs. Each plant would insert their plant specific data, however they would use the pattern document's approach for addressing the process and technical issues.

V. USE OF EXISTING PROGRAMS: THE MAINTENANCE RULE AND BWRVIP

Compliance with regulatory and/or code requirements is the basis for many of the current plant maintenance programs for structures and components within the scope of the license renewal rule. Other programs have evolved through significant industry efforts to resolve operating experience issues. The objectives, scope and procedures for these programs typically focus on establishing the current condition of the structure or component and providing information for timely corrective action, when necessary. An important point is that the basis and content (techniques, processes, etc.) of these types of programs is independent of the period of operation. Therefore, the need to demonstrate that these programs are effective in a license renewal application is not purposeful. The exercise would only result in "re-presenting" information the NRC has already reviewed and accepted as part of the resolution of, or compliance with, an associated regulatory issue/requirement. The more practical and meaningful approach would be just listing these types of programs in the application with a discussion showing that the program manages the aging effect. A commitment would then be made to continue these programs through out the extended operating period.

Two examples of these types of programs are: (1) the structural monitoring program that results from compliance with Maintenance Rule, and (2) the application of the BWR Vessel Internals Program (VIP) inspection guidance.

The Plant Hatch structural monitoring program that was established for compliance with the Maintenance Rule has evolved into a "full scope" condition-monitoring program. Qualified personnel perform comprehensive, walk-downs of structures to record, categorize and disposition the observed conditions. Periodic inspections performed as part of previously existing programs (e.g., structures settlement monitoring, soil and ground water analysis, buried component assessment, ASME Section IWE, etc.) are also made part of the overall structural monitoring program. The information is recorded in a manner to facilitate ongoing monitoring. Industry and regulatory guidance has been adopted to provide substantial direction with regard to relevant aging effects, susceptible locations, manifestations and precursors to recognize deterioration and grade the as-found conditions. Acceptance criteria are established and used in the grading of the structures and to determine when it is necessary to establish goals to restore the condition of a structure. The NRC has reviewed the program as part of their baseline maintenance rule compliance inspections. Guidance has also been provided to the resident and region-based inspectors for their ongoing verifications. Therefore, in view of these considerations, utilities should only have to commit to continue the structural monitoring program as the basis for providing reasonable assurance that aging will not prevent the structures from performing their intended functions. SNC believes that no new programs or significant enhancements to the current program for Plant Hatch will be necessary. SNC expects to use the review of the intake structure, as part of the teaming program, to establish that this approach complies with the requirements of the license renewal rule.

The BWRVIP is developing a comprehensive program that allows utilities with BWR plants to manage degradation associated with the reactor vessel and internals components. Cost-effective evaluation methodologies, inspection techniques and repair alternatives are being defined and qualified. Each operating plant can select the appropriate alternative. It has been recognized by the industry that the goals, scope of evaluations and inspection requirements established by the BWRVIP also apply to the extended operating period. A license renewal appendix has been developed for each of the vessel internal, component groupings. The appendix demonstrates how the inspection guidance complies with the License Renewal Rule. In most cases, this involves continued use of the current operating term techniques. In a few cases, the analysis developed by the BWRVIP was based on 40-year time limited assumptions and it was necessary to demonstrate that these analyses could be extended to 60 years. The appendix also identifies any issues requiring plant-specific reviews or evaluations. Full acceptance by the NRC of this approach is needed in the near term. It would not be cost effective for SNC to provide a plant-specific demonstration that the generic inspection guidance will provide reasonable assurance that the affects of aging are effectively managed for the renewal term. The technical resources that are currently working on the BWRVIP can effectively disposition the issues affecting the acceptance of the generic guidance for license renewal. Near-term application of these license renewal appendices is planned for Plant Hatch. This will facilitate early interaction with the NRC and timely identification and assessment of the potential impact of any compliance issues.

VI. CONCLUSION

SNC has decided to proceed with developing a license renewal application for Plant Hatch. The final decision on whether to submit the application will be made once the application is completed. The program strategies, and participation in teaming relationships, are expected to provide opportunities for early interaction with the NRC. The goal is to resolve the areas of uncertainty affecting implementation of the license renewal rule and to assess the impact of the resolutions on the Plant Hatch program. The Plant Hatch teaming initiatives will also provide license renewal technology to the BWR fleet, and serve to provide an oversight role during the implementation of the Plant Hatch program.

Lewis Sumner
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SOUTHERN COMPANY
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April 13, 1998

HL-5609

Docket Nos. 50-321
50-366

Dr. S. A. Jackson, Chairman
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001

Subject: Baltimore Gas & Electric Company's License Renewal Application for Calvert Cliffs Nuclear Power Plant

Dear Chairman Jackson:

Southern Nuclear Operating Company (SNC) supports the efforts of the Commission regarding the subject of license renewal and resolution of the policy and process issues that will lead to a renewed operating license.

The principal objective the Commission set forth, in promulgating the existing Part 54 rule was to develop a predictable, stable, and efficient license renewal process that assures public health and safety. The recent announcement by Baltimore Gas and Electric Company (BG&E) that it would submit a formal application for a renewed license for the Calvert Cliffs Nuclear Power Plant represents the first application for a renewed license under 10 CFR Part 54. SNC applauds and supports BG&E's decision to submit an application as a significant contribution towards development of a license renewal process that is predictable, stable, and efficient.

In addition, SNC expects that generic issues will continue to be resolved through NEI/NRC staff interactions. These interactions are necessary in order to define the process which meets the stated objective - a license renewal process that can be implemented in a timely and effective manner by any licensee.

R/3

SNC plans to complete work on a license renewal application by the end of 1999 (Ref. 1). In addition, SNC actively participates in NEI license renewal activities. As indicated in Reference 1, SNC intends to work in support of the previously stated objective through both plant-specific and generic interactions with the NRC staff.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

HLS/RDB

References:

1. SNC letter HL-5503, H. Lewis Sumner, Jr. to U. S. Nuclear Regulatory Commission, Document Control Desk, "Edwin I. Hatch Nuclear Plant Submittal Plan for License Renewal Technical Information and Request for Waiver of Review Fees," October 27, 1997.

cc: Southern Nuclear Operating Company
Mr. P. H. Wells, General Manager - Plant Hatch
Mr. H. L. Sumner, Jr., Vice President - Plant Hatch

U. S. Nuclear Regulatory Commission, Washington, DC
Mr. L. N. Olshan, Project Manager - Hatch
Mr. C. Grimes, Director, NRC License Renewal Directorate
Mr. Raj Anand, NRC License Renewal Project Manager - Hatch

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. B. L. Holbrook, Senior Resident Inspector - Hatch

Nuclear Energy Institute
Mr. D. J. Walters

NRC/SNC Meeting

August 18, 1998

Plant Hatch
License Renewal Process

Meeting Agenda

☐ **INTRODUCTIONS**	*Charles Pierce*
☐ OVERVIEW - HATCH LICENSE RENEWAL PROGRAM	*Louis Long/Ray Baker*
☐ HATCH LICENSE RENEWAL PROCESS DOCUMENT	*All*
☐ APPLICATION DOCUMENTS SCHEDULE	*Ray Baker*
☐ ELECTRONIC DOCKET	*All*
☐ SUMMARY/CONCLUSION	*All*

8/4

Meeting Agenda

☐ INTRODUCTIONS	*Charles Pierce*
☐ **OVERVIEW**	*Louis Long*
- HATCH LICENSE RENEWAL PROGRAM	*Ray Baker*
☐ HATCH LICENSE RENEWAL PROCESS DOCUMENT	*All*
☐ APPLICATION DOCUMENTS SCHEDULE	*Ray Baker*
☐ ELECTRONIC DOCKET	*All*
☐ SUMMARY/CONCLUSION	*All*

Overview

Hatch License Renewal Program

Louis B. Long - Vice President, Technical Services

Overview

Hatch License Renewal Program

Louis B. Long - Vice President, Technical Services
Ray Baker - Hatch License Renewal Project Manager

Overview - Hatch License Renewal Program

Industry Participation

- ✓ NEI License Renewal Working Group
- ✓ NEI License Renewal Implementation Task Force
- ✓ EPRI Life-Cycle Management Subcommittee
- ✓ BWR Owners Group License Renewal Committee
- ✓ BWR VIP
- ✓ WOG Life-Cycle Management Committee

Overview - Hatch License Renewal Program

- ✓ Industry Participation

Program Announcement

- ✓ Announcement Letter - October 27, 1997

Program Announcement

- ✓ Announcement Letter - October 27, 1997

  - ➢ Application Documents Development

Program Announcement

- ✓ Announcement Letter - October 27, 1997
  - ➢ Application Documents Development
  - ➢ Participate with NRC and Industry in Issues Resolution

Overview - Hatch License Renewal Program

- ✓ Industry Participation
- ✓ Program Announcement

Initiatives

- Hatch License Renewal Process Document
- Collaborative Efforts with PECO Energy and Northern States Power

Overview - Hatch License Renewal Program

✓ Industry Participation
✓ Program Announcement
✓ Initiatives

Program Statistics

- Resources:
  - Budget: $13.2 M over 5 years (development cost, not including NRC fees and hearings)
  - Staffing: More than 20 equivalent full-time personnel
    - Project Team - 18 (16 full-time, 2 half-time)
    - Consultants, Contractors - 3+ (equivalent)
      - Tetra Tech NUS, Georgia Power Company, LCM Engineering, GE Nuclear, T.Heroux Consulting, Winston & Strawn, Balch & Bingham, and others

Program Statistics
(continued)

- Schedule:
  - Application Documents Development: Complete by December 31, 1999
  - Go-ahead Decision: 1st Quarter, 2000

★If authorized by co-owners, submit application 1st Quarter, 2000

Program Statistics
(continued)

- Major Tasks Status:
  - Scoping: **Complete**
  - Establish Eval. Boundaries: **Complete**
  - Screening: **Near Completion**
  - Aging Management Review Prerequisites:
    - Identify Relevant Aging Effects **Started**
    - Identify Aging Mgmt. Activities **Started**
  - Aging Management Reviews **Not Started**

Program Statistics
(continued)

- Major Tasks Status (continued):
  - Identify Time-Limited Aging Analyses: **In Progress**
  - Evaluate Time-Limited Aging Analyses: **In Progress**
  - Identify Exemptions: **Complete**
  - Environmental Activities: **In Progress**
  - Application Development: **Started**
  - FSAR Supplement **Not Started**

Overview - Hatch License Renewal Program

✓ Industry Participation
✓ Program Announcement
✓ Initiatives

★Program Statistics
Program Elements

Program Elements

- Scoping Results
  - Functions in scope — **106**
    - Safety-related — 79
    - Non-safety related — 8
    - ATWS Rule — 13
    - Fire Protection Rule — 51
    - EQ Rule — 26
    - Station Blackout Rule — 33
  - Functions not in scope — **169**

Program Elements

- Evaluation Boundaries
  - 142 separate evaluation boundaries produced
  - Mechanical boundaries are being "overlayed" electronically on existing plant P&Ids
  - Mechanical boundaries identify interfaces with structures (both piping support and equipment support interfaces)
  - Electrical components evaluated on a plant "spaces" basis

Program Elements

- Screening
  - Group components into commodities within Evaluation Boundaries
  - Make "active/passive" determination
  - Make "long-lived" determination
  - Identify how component/commodity supports intended functions

Program Elements

- AMR Prerequisites
  - Systematically identify relevant aging effects
    - Mechanical: use a standard set of tools to identify aging effects. The tools use commodity materials and environment information, based on an exhaustive review of industry literature, including NRC generic communications.

Program Elements

- Systematically identify relevant aging effects (continued)
  - Structures: use the industry Class I Structures report and other industry reports plus a review of generic communications to identify aging effects.
  - Electrical: use the Cable AMG and other industry reports plus a review of generic communications.
- Systematically identify plant aging management activities

Program Elements

- Aging Management Reviews
  - Map the aging effects and aging management activities for each commodity
  - Evaluate plant-specific commodity maintenance history
  - Identify the set of activities to credit during period of extended operation to adequately manage aging effects (can be existing activities, enhancements, or new activities, or a combination)

Program Elements

- Identify Time-Limited Aging Analyses
  - Review docket correspondence summaries for keywords that may indicate potential applicability
  - Screen A/E calculations for potential applicability
  - Review A/E calculations that pass first screening criterion for remaining five criteria (two step approach to screening calculations)

Program Elements

- Evaluate Time-Limited Aging Analyses
  - Identify disposition process for each TLAA
  - Establish disposition/resolution schedule

Program Elements

- Identify Exemptions
  - Review all exemptions specified in Operating Licenses for TLAA relationships

★No exemptions met criteria

Program Elements

- Environmental Activities
  - Report Development
    - Report Outline developed
    - Survey Plan developed
    - collecting environmental data
  - Evaluating internal analyses with regard to proposed SAMA requirements in Draft Reg. Guide
  - Seeking input from State environmental agencies for Environmental Report

Hatch License Renewal Program

★Fully Funded

★Fully Staffed

★Full Scope

★Fully Engaged

Meeting Agenda

- ☐ INTRODUCTIONS — *Charles Pierce*
- ☐ OVERVIEW - HATCH LICENSE RENEWAL PROGRAM — *Louis Long/Ray Baker*
- ☐ **HATCH LICENSE RENEWAL PROCESS DOCUMENT** — *All*
- ☐ APPLICATION DOCUMENTS SCHEDULE — *Ray Baker*
- ☐ ELECTRONIC DOCKET — *All*
- ☐ SUMMARY/CONCLUSION — *All*

Process Document

- ***Discuss NRC/NEI issue resolution process and relationship of Hatch Process Document to the industry process***
- ***Discuss forum for resolution of specific issues addressed by Process Document in context of NEI database of SRP comments***
- ***Discuss desired schedules for resolution of issues not contained in NEI database of SRP comments***
- ***Feedback on priorities***

NRC/NEI Generic Issues Resolution Process

- Re-engagement of industry and NRC on issues is viewed as a very positive development
- New structure for issues resolution within NRC is encouraging

★ **Southern Company is strongly commited to making the generic issues resolution process succeed**

Process Document

- ***Discuss NRC/NEI issue resolution process and relationship of Hatch Process Document to the industry process***
- ***Discuss forum for resolution of specific issues addressed by Process Document in context of NEI database of SRP comments***
- ***Discuss desired schedules for resolution of issues not contained in NEI database of SRP comments***
- ***Feedback on priorities***

Process Document Issues / NEI Comments Database Comparison

	Area	NEI SRP Comment	Recommendation
1.	Functional-based approach for scoping	No Comment	Discuss philosophy separately with NRC
2.	Use of intended function at system and component levels	Yes (SRP comment 82)	Follow resolution of issue with NEI and NRC. SNC implementation appears consistent with BG&E and Duke
3.	Practical considerations in limiting plant review	No comment	Discuss philosophy separately with NRC
4.	Screening of structures performed by HNP process	No comment	Discuss philosophy separately with NRC
5.	Commodity groups established during screening	Yes (SRP comments 19, 67, 70)	Follow resolution of issue with NEI and NRC. Need earlier resolution than currently scheduled to minimize rework (late-1998).
6.	Consumables are not in scope of review	Yes (SRP comments 25, 62, 127, 138, 162)	Follow resolution of issue with NEI and NRC. Expect timetable to be sufficient (mid-1999).
7.	Active complex assemblies are dispositioned from AMR	Yes (SRP comment 2)	Follow resolution of issue with NEI and NRC. Expect timetable to be sufficient (mid-1999)
8.	Review process for existing programs	Yes (SRP comments 161, 116, 39, etc.)	Follow resolution of issue with NEI and NRC. Expect timetable to be sufficient (mid-1998).

Process Document

- ***Discuss NRC/NEI issue resolution process and relationship of Hatch Process Document to the industry process***
- ***Discuss forum for resolution of specific issues addressed by Process Document in context of NEI database of SRP comments***
- ***Discuss desired schedules for resolution of issues not contained in NEI database of SRP comments***
- ***Feedback on priorities***

Process Document

- *Discuss NRC/NEI issue resolution process and relationship of Hatch Process Document to the industry process*
- *Discuss forum for resolution of specific issues addressed by Process Document in context of NEI database of SRP comments*
- *Discuss desired schedules for resolution of issues not contained in NEI database of SRP comments*
- *Feedback on priorities*

Meeting Agenda

☐ INTRODUCTIONS	*Charles Pierce*
☐ OVERVIEW - HATCH LICENSE RENEWAL PROGRAM	*Louis Long/Ray Baker*
☐ HATCH LICENSE RENEWAL PROCESS DOCUMENT	*All*
☐ **APPLICATION DOCUMENTS SCHEDULE**	*Ray Baker*
☐ ELECTRONIC DOCKET	*All*
☐ SUMMARY/CONCLUSION	*All*

Schedules

- Preliminaries
  - Intake Structure — 3rd/4th Qtr
  - Recirculation System Piping — 3rd/4th Qtr
  - SLC — 4th Qtr
  - Selected BWRVIP Products — 4th Qtr/1st Qtr
- Application Documents/Products — 1999/2000

Meeting Agenda

☐ INTRODUCTIONS	*Charles Pierce*
☐ OVERVIEW - HATCH LICENSE RENEWAL PROGRAM	*Louis Long/Ray Baker*
☐ HATCH LICENSE RENEWAL PROCESS DOCUMENT	*All*
☐ APPLICATION DOCUMENTS SCHEDULE	*Ray Baker*
☐ **ELECTRONIC DOCKET**	*All*
☐ SUMMARY/CONCLUSION	*All*

Electronic Docket

- Technical Issues
- Regulatory Issues

Meeting Agenda

☐ INTRODUCTIONS	*Charles Pierce*
☐ OVERVIEW - HATCH LICENSE RENEWAL PROGRAM	*Louis Long/Ray Baker*
☐ HATCH LICENSE RENEWAL PROCESS DOCUMENT	*All*
☐ APPLICATION DOCUMENTS SCHEDULE	*Ray Baker*
☐ ELECTRONIC DOCKET	*All*
☐ **SUMMARY/CONCLUSION**	*All*

Summary/Conclusion

- Recap - open items
- Final thoughts
- Adjourn

Raymond D. Baker
Southern Nuclear
Hatch License Renewal
Project Manager

April 13, 1999

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Jeff Mulvehill
Southern Nuclear
Hatch License Renewal
Project Senior Engineer

April 13, 1999

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Overview - Chapter Headings

General Section

Exhibit A Technical Section

1.0 Introduction and Description of Hatch Nuclear Plant

2.0 IPA Methodology

3.0 Generic Safety Issues

4.0 SSCs Subject to an Aging Management Review

5.0 Identification and Management of Aging Effects

6.0 Aging Effects

Exhibit B Updated Final Safety Analysis Report Supplement

Exhibit C Technical Specification Changes

Exhibit D Proposed Operating Licenses

Exhibit E Environmental Report

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Methodology

2.0 IPA Methodology
- 2.1 General
- 2.2 Scoping
- 2.3 Boundaries
- 2.4 Screening
- 2.5 TLAAs

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Process Mapping

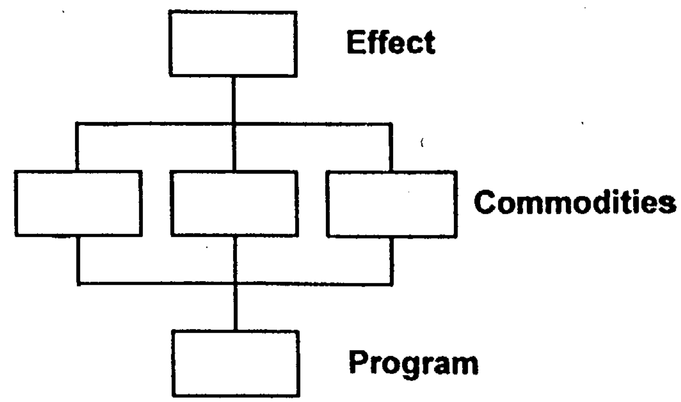
Systems

Intended Functions

Functional Boundaries

Screened Commodities

System-by-System



Plant-wide

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Generic Safety Issues

3.0 Generic Safety Issues
- 3.1 Evaluation
- 3.2 GSI-XXX
- 3.3 GSI-YYY
- 3.4 References for Section 3

- Each GSI section addresses
  - Background
  - Generic Industry Technical Rationale
  - Hatch Confirmatory Research
  - Hatch License Renewal Position for GSI

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Systems and Structures Subject to an AMR

- 4.0 SSCs Subject to an Aging Management Review
  - 4.1 Structures
    - 4.1.1 Structure XXX
      - 4.1.1.1 Description
      - 4.1.1.2 Intended Functions
      - 4.1.1.3 Function Boundaries
      - 4.1.1.4 SSCs Subject to Aging Management Review
      - 4.1.1.5 Component Functions
    - 4.1.2 Structure YYY

      (Same format as 4.1.1)
  - 4.2 Reactor

    (Same format as 4.1.1, etc.)
  - 4.3 Reactor Coolant System and Connected Systems

    (Same format as 4.1.1, etc.)

Systems and Structures Subject to an AMR (Continued)

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4.4 Engineered Safety Features
(Same format as 4.1.1, etc.)

4.5 Instrumentation and Controls
(Same format as 4.1.1, etc.)

4.6 Electric Power
(Same format as 4.1.1, etc.)

4.7 Auxiliary Systems
(Same format as 4.1.1, etc.)

4.8 Steam and Power Conversion System
(Same format as 4.1.1, etc.)

4.9 Radioactive Waste
(Same format as 4.1.1, etc.)

Systems and Structures Subject to an AMR (Continued)

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- 4.10 Electrical Components
  - 4.10.1 Electrical Components Subject to Aging Management Review
  - 4.10.2 Discussion of Special Topics
  - 4.10.3 Structures and Areas Containing Electrical Components Subject to Aging Management Review
  - 4.10.4 Aging Management Review of Electrical Components Using Spaces Approach
  - 4.10.5 References
  - Table 4.10-1 Electrical Component Types Subject to an Aging Management Review and Their Intended Functions

Identification and Management of Aging Effects

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5.0 Identification and Management of Aging Effects

- 5.1 Commodity Aging Effects Terminology
- 5.2 Commodity Groups
  - 5.2.1 Commodity Group 1
    - 5.2.1.1 Description (including materials, internal and external environments)
    - 5.2.1.2 Plausible and Detrimental Aging Effects
    - Table 5.2-xx Plausible and Detrimental Aging Effects for Commodity Group 1
  - 5.2.2 Commodity Group 2
    - (Repeat for each commodity group)
- 5.3 Management of Detrimental Aging Effects
  - (subdivided by aging effect)
  - (Demonstration is made for each aging effect)

6.0 Aging Effects

- 6.1 Aging Effect xxx
- 6.2 Aging Effect yyy

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Aging Management Programs

- Aging Management Programs are described in the FSAR Supplement
- Programs are grouped into three broad areas
  - Monitoring Programs
  - Mitigating Programs
  - Supporting Programs

Lewis Sumner
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Hatch Project Support

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SOUTHERN COMPANY
Energy to Serve Your World℠

May 14, 1999

Docket Nos. 50-321
50-366

HL-5763

Mr. John C. Hoyle, Secretary
U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Edwin I. Hatch Nuclear Plant
Recirculation System Pressure Boundary Licensing Report
Product Submittal to NRC

Dear Ladies and Gentlemen:

Southern Nuclear Operating Company (SNC) is submitting 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. The report was generally developed using the process described in our April 13, 1998, letter to the NRC, "License Renewal Process Methodology Document." We expect the format of this report to be similar to the format that will be followed in the application; however, it may be organized differently for a full application.

This licensing report was developed under a collaborative effort with PECO Energy and Northern States Power. The goal of this collaborative effort was to develop a report format that all three licensees could use to submit similar information without significant revision, should they elect to pursue an application. This report presents the technical information formatted specifically for Plant Hatch. Similar licensing reports by other licensees developed from the same technical information would tend to reduce both utility and NRC resource requirements.

SNC is planning for its license renewal application, if submitted, to be a first-of-a kind electronic docket submittal. This licensing report also represents an example of this type of submittal. The hard-copy contains underlined words which represent links to other portions of

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the document. The enclosed diskette contains the electronic version of this licensing report in HTML format.

SNC requests the NRC review the enclosed report for its level of detail and technical content with a view that SNC would use this report as a template for the technical material related to Class 1 piping systems in the application, if submitted. SNC believes that the enclosed recirculation system licensing report meets the information requirements of 10 CFR 54.21(a), "Contents of application-technical information," and provides an adequate level of detail for the demonstration required by 10 CFR 54.29(a)(1), "Standards for issuance of a renewed license." SNC would like to receive any "requests for additional information" concerning this report by June 1, 1999 and resolve them by August 1, 1999 in order to support ongoing internal activities toward development of an application by the end of 1999.

Respectfully submitted,

Lewis Sumner

H. L. Sumner, Jr.

Enclosures:
Reactor Recirculation System Pressure Boundary Components
Diskette with HTML version of above report

HLS/JEH

cc Southern Nuclear Operating Company
Mr. P. H. Wells, General Manager - Plant Hatch

U. S. Nuclear Regulatory Commission, Washington, DC
Mr. L. N. Olshan, Project Manager - Hatch

U. S. Nuclear Regulatory Commission, Region II
Mr. L. A. Reyes, Regional Administrator
Mr. J. T. Munday, Senior Resident Inspector – Hatch

Edwin I. Hatch Nuclear Plant, Units 1 and 2

Reactor Recirculation System Pressure Boundary Components

Demonstration of Compliance
With The Requirements Of 10 CFR 54,
the License Renewal Rule

Table of Contents

Tables

Figures

1.0 INTRODUCTION

This Recirculation System Report presents information required by §54.21[a] and [c], and §54.22 of the License Renewal Rule (Ref. 1), which is hereafter referred to as the "Rule." The report is a special product as discussed in SNC correspondence with the NRC dated January 15, 1998. Consequently, the scope of this report is narrowly focused. The report only addresses the pressure boundary for the Edwin I. Hatch Nuclear Plant (Plant Hatch) Units 1 and 2, Reactor Recirculation System (RRS). The report was produced using a preliminary version of the Plant Hatch process methodology (Ref. 2) for implementing the requirements of the Rule that was submitted to the NRC in April 1998. However, a detailed discussion of this methodology for identifying systems, structures, and components within the scope of the Rule and subject to an aging management review is beyond the scope of this report. This report also utilizes additional guidance from various NRC and industry documents generated subsequent to submittal of the Process Document. The process used by Plant Hatch in producing this document is consistent with the lessons learned from industry communications with the NRC staff and the generic guidance in NEI 95-10, "Revision 0, Industry Guideline on Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," March 1996 (Ref. 3).

2.0 DESCRIPTION OF THE REACTOR RECIRCULATION SYSTEM AND ITS INTENDED FUNCTION

The Edwin I. Hatch Nuclear Plant (Plant Hatch) Units 1 and 2, Reactor Recirculation System (RRS) provides the driving flow for the reactor vessel jet pumps, which in turn provide reactor coolant flow and reactor power control. These components form a portion of the reactor coolant pressure boundary.

The RRS consists of two parallel loops, each consisting of a recirculation pump, suction and discharge block valves, piping, fittings, flow elements, and connections supporting flow, differential pressure, and temperature instrumentation. The system functions and requirements are described in the Edwin I. Hatch Nuclear Plant Unit 1 Final Safety Analysis Report (FSAR), Chapter 4 and Unit 2 FSAR, Chapter 5. The RRS interfaces with the Residual Heat Removal (RHR) and Reactor Water Cleanup (RWCU) systems to provide a flow-path in support of shutdown cooling, emergency low-pressure coolant injection (LPCI), RWCU and reactor water level control functions. The system is part of the reactor coolant pressure boundary. Therefore, it also functions to maintain the pressure boundary during normal operation, transients, and accident scenarios to prevent the release of radioactive liquid and gas.

The internal environment of the RRS pressure boundary is reactor water, at approximately 533°F and 1055 psia during normal plant operation. Water quality is maintained within specified limits. During plant conditions that require the operation of the shutdown cooling mode of RHR, reactor water can be cooled to approximately 117°F via the RHR heat exchangers and recirculated back to the reactor through the RRS piping. During plant shutdown conditions, the water temperature in this piping can be as low as 70°F. Hydrogen water chemistry (HWC) limits the RRS oxygen content to less than 5 ppb. The external environment for the system is relatively non-aggressive because it is inerted with nitrogen during operation.

The design specification and applicable regulations required that these components be designed for pressure, thermal, dead and live loads, seismic, pipe break loads, and fracture prevention. Various piping stress reports identify the applied loads, load combinations, and safety factors that need to be considered to verify that operating conditions are consistent with the current licensing basis (CLB).

The Process Document (Ref. 2) was used to identify the RRS functions that are within the scope of the Rule (Ref. 1). The scoping process determined that the following functions are intended functions as defined by the Rule, and that the RRS is required to accomplish these intended functions:

- Maintain the reactor coolant pressure boundary integrity.
- Trip the Recirculation Pump Trip (RPT) breakers.

As stated in Section 1.0, this report only addresses the intended function of maintaining the reactor coolant pressure boundary integrity. The function of tripping the RPT breakers is beyond the scope of this report, but will be addressed in the renewal application for Plant Hatch, if submitted.

3.0 REACTOR RECIRCULATION SYSTEM PRESSURE BOUNDARY COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW

An initial review of the system was performed to identify the components that require an aging management review. These components were grouped into commodities to facilitate performance of the aging management reviews. The process document (Ref. 2) refers to this initial review as the "screening process." The results from applying the screening process to the RRS are described in this section. The second part of the integrated plant assessment (IPA), the aging management review of systems, structures, and components subject to aging management review, is described in Section 4.0 of this document.

3.1 Evaluation Boundaries

The evaluation boundaries for the RRS pressure boundary components are shown on Figure 1 for Unit 1 and Figure 2 for Unit 2. The system begins at the pipe weld connection to the reactor vessel nozzle/safe end assembly and includes the suction piping, suction valve body/bonnet, pump casing/cover, discharge valve body/bonnet, and discharge piping up to the weld connections to the reactor vessel nozzle/safe end assemblies. The connecting welds are part of the RRS. The associated flow venturis, drain/vent piping and valve bodies, and instrument piping and valve bodies are included in the evaluation boundary. Interfacing systems, piping and equipment supports, pipe whip restraints, pump/motor pedestal supports, electro-mechanical components and containment penetrations are not included in this review. Interfacing systems, structures and components will be addressed in the renewal application.

Figure No. 1
Unit 1 Reactor Recirculation Evaluation Boundary Diagram

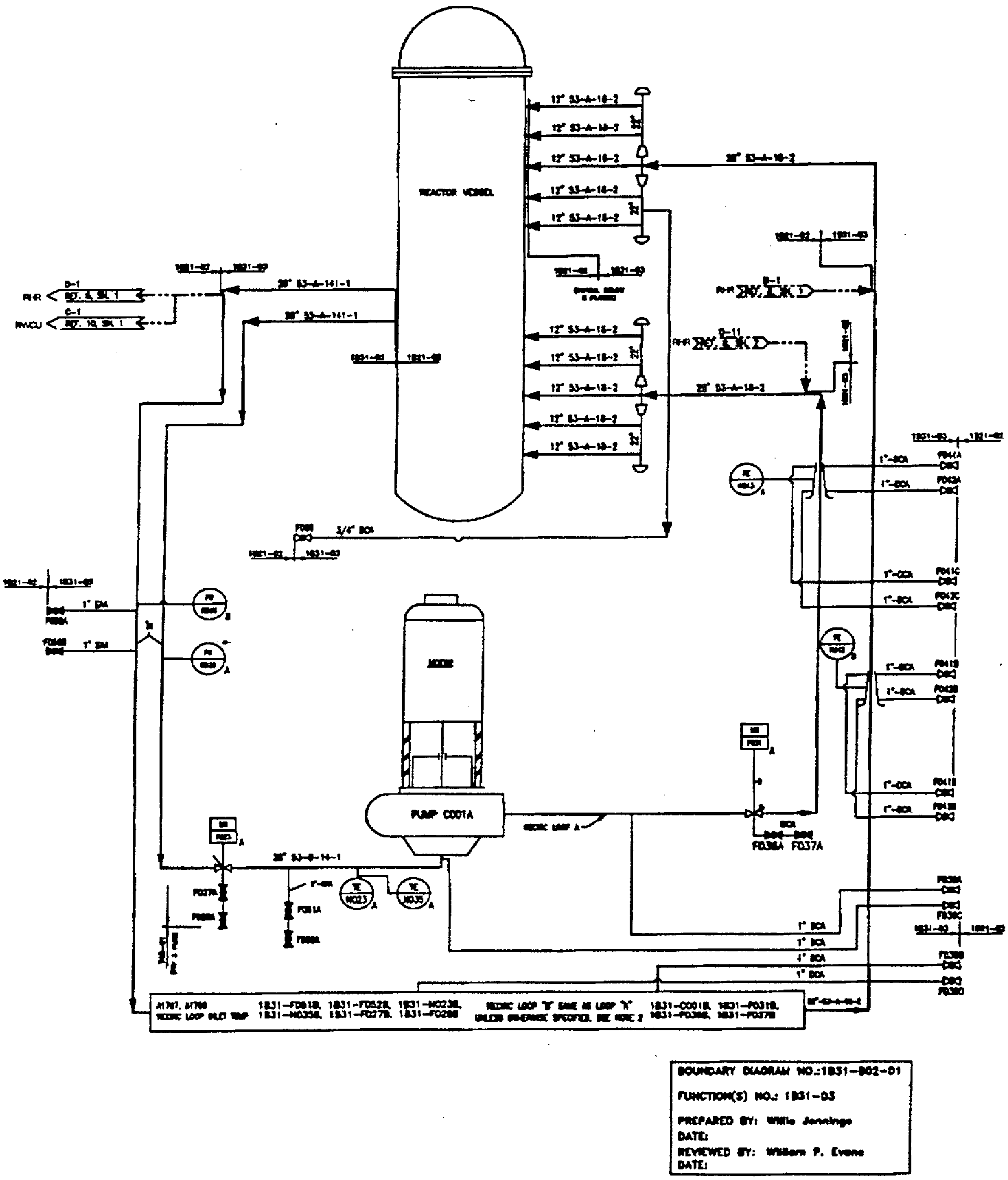
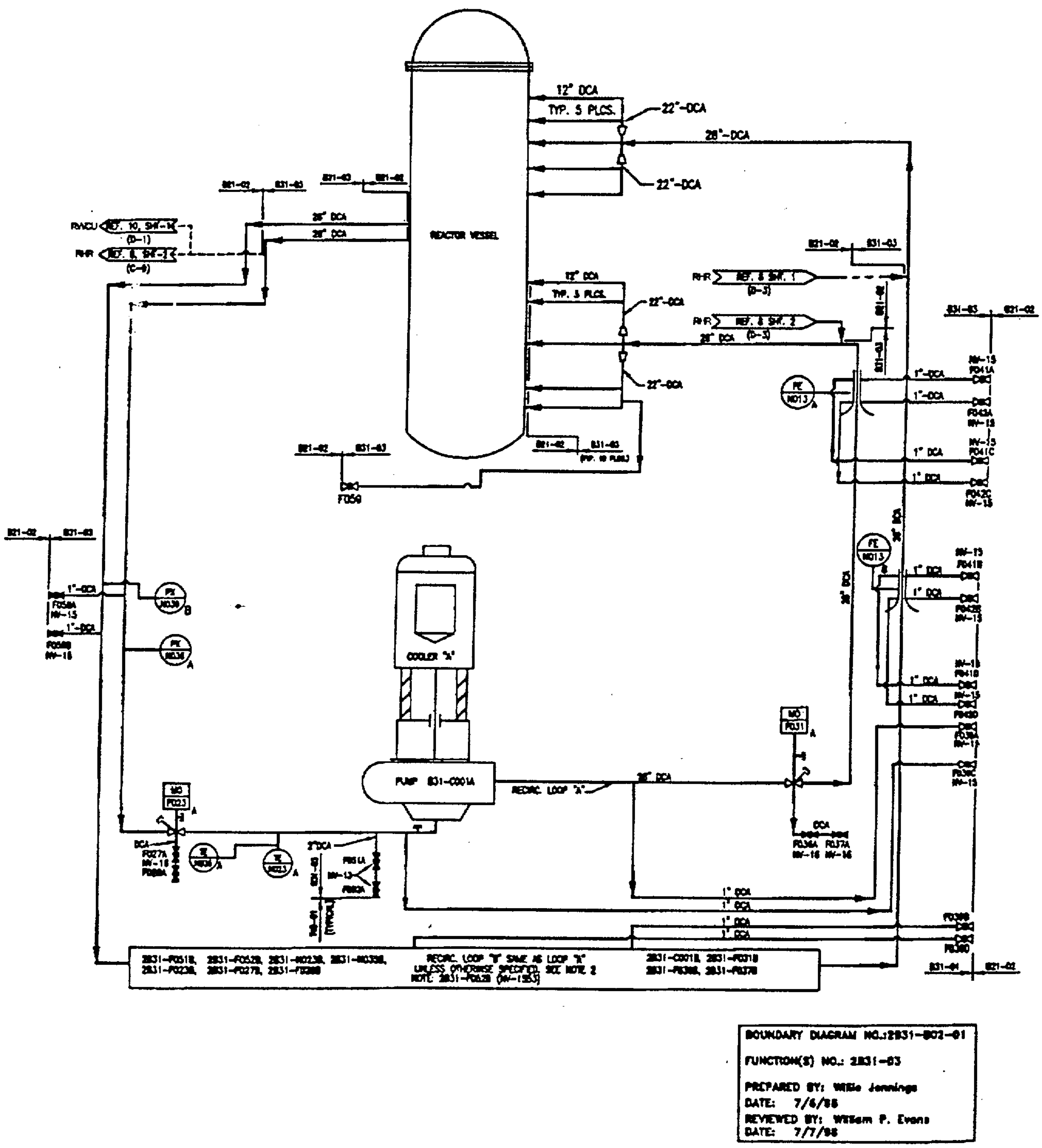


Figure No. 2
Unit 2 Reactor Recirculation Evaluation Boundary Diagram



3.2 Components Subject to Aging Management Review

Section 54.21(a)(1) of the Rule (Ref. 1) provides the requirements for identifying the RRS pressure boundary components within the evaluation boundary that are subject to aging management review. For this system, all of the pressure boundary components, with the exception of the recirculation pump seals, are passive, long-lived, and subject to aging management review. These components are listed in Table 1. The qualified life and replacement frequency for the recirculation pump seals are less than 40 years. Therefore, the seals are short-lived components and not subject to aging management review.

The pressure boundary components are grouped, as shown below, by type, material and/or size. These commodity groupings allow a more efficient review of the aging effects and facilitate the aging management review of some of the component groups.

- Stainless Steel Large-bore Piping
- Stainless Steel Small-bore Piping and Valve Bodies
- Pump Casings
- Large-bore Valve Bodies
- Integral Attachments
- Pressure Boundary Bolting

Table 1
Reactor Recirculation System
Components Reviewed and Evaluated in this Report

System Component	Component Boundary	Commodity Grouping
Recirculation Pumps	Casing and Cover Casing Bolts/Studs	<u>Pump Casings</u> <u>Pressure Boundary Bolting</u>
Discharge/Return Motor Operated Valves	Body and Bonnet Bonnet to Body Bolting	<u>Large-bore Valve Bodies</u> <u>Pressure Boundary Bolting</u>
Suction/Inlet Motor Operated Valves	Body and Bonnet Bonnet to Body Bolting	<u>Large-bore Valve Bodies</u> <u>Pressure Boundary Bolting</u>
Class 1 Large-bore Suction Piping	28 inch Piping from the Vessel Recirculation Suction Nozzle to the Pump Suction Nozzle Decontamination Connection	<u>Stainless Steel Large-bore Piping</u>
Class 1 Large-bore Discharge Piping	28 inch Piping from the Pump Discharge Nozzle through the 12 inch Risers to the Vessel Return Nozzles Flow Venturis Outer Housing Decontamination Connection	<u>Stainless Steel Large-bore Piping</u>
Integral Attachments	Welded Integral Attachments	<u>Integral Attachments</u>
Small-bore Piping	2 inch and under piping connected to large-bore suction and discharge piping for vents, drains, instruments (up to and including the first isolation valve), etc. Thermowells Flow Venturis Pressure Taps	<u>Stainless Steel Small-bore Piping and Valve Bodies</u>
Small-bore Manual Valves	Valve Body and Bonnet	<u>Stainless Steel Small-bore Piping and Valve Bodies</u>

3.3 Component Functions

Each of the components that make up the six commodity groups discussed in this report has two functions that support the intended function of maintaining the reactor coolant pressure boundary integrity. Those component functions are:

- Provide pressure boundary or fission product retention barrier to protect public health and safety in the event of any postulated design basis events.
- Provide a pressure retaining boundary so that sufficient flow and adequate pressure are delivered.

4.0 MANAGEMENT OF AGING EFFECTS FOR LICENSE RENEWAL

Section 54.21(a)(3) of the Rule (Ref. 1) requires an aging management review to demonstrate that the aging effects will be adequately managed so that the intended functions will be maintained consistent with the CLB for the extended period of operation. This section identifies:

a) the detrimental aging effects for the RRS components that are subject to aging management review, as determined in Section 3.0 of this report, and
b) aging management activities associated with the components.

4.1 Commodity Aging Effects Terminology

The plausible aging effects are identified and reviewed in this section. Available industry literature was reviewed to identify the plausible aging effects for the materials and environments applicable to RRS components. Industry operating experience, NRC generic communications, EPRI Report TR-103843 (Ref. 5), and Plant Hatch operating experience were also reviewed to ensure that all plausible aging effects for the relevant material/environment combinations have been identified.

Plausible aging effects that are determined to be applicable for Plant Hatch and that could cause sufficient degradation of the Structure, Component, or Commodity (SCC) to potentially result in a loss of the intended function are identified as detrimental aging effects requiring aging management. This process resulted in the listing of plausible and detrimental aging effects shown in Tables 2 through 5.

4.2 Commodities

The RRS components subject to aging management review, for the commodity groups identified in Section 3.2, are further discussed in the following sections.

4.2.1. Stainless Steel Large-bore Piping

Commodity Description

The RRS primary process piping is large-bore stainless steel piping, ranging from 12 inch to 28 inch. This commodity group also includes the outer housing of the flow venturis and decontamination connections.

The Unit 1 piping is designed to ANSI B31.7, Class 1, 1969 Edition and is made of Type 304 stainless steel. In response to concerns with intergranular stress corrosion cracking (IGSCC), a combination of Induction Heating Stress Improvement (IHSI) and Mechanical Stress Improvement Process (MSIP) was used to limit additional cracked weldments, and weld overlays were used to repair crack weldments. The weld overlays are made of ER308L grade stainless steel, an IGSCC resistant material. The filler metal for weld overlay repair of Inconel to stainless steel welds is Inconel 82, also an IGSCC resistant material. All weld overlay designs were prepared assuming a bounding 360-degree circumferential, crack-oriented through-wall flaw in accordance with the requirements of the "Standard Weld Overlay" design intended for indefinite service per NUREG-0313 (Ref. 4).

The Unit 2 piping was originally designed to ASME Section III, Class 1, 1971 Edition. In response to concerns with IGSCC, the large-bore piping was upgraded in 1984 from 304 stainless steel to 316NG stainless steel, which is more resistant to IGSCC. The replacement piping was designed and furnished to the ASME Section III, 1980 edition including addenda through winter 1981.

For both units, the internal environment is reactor water and the external environment is the drywell environment, which is benign. These environments are more completely described in Section 2.0 of this report.

Aging Effects

Plausible and Detrimental Aging Effects for Stainless Steel Large-bore Piping Components are listed in Table 2.

Crack initiation and growth due to thermal fatigue is described in Section 5.1. Crack initiation and growth due to thermal fatigue of the Class 1 stainless steel piping system is a detrimental aging effect for the large-bore piping and will be managed by the aging management program described in Section 4.3.1. Thermal fatigue is a time-limited aging analysis (TLAA) issue and is evaluated in Section 7.2.

Crack initiation and growth due to IGSCC is described in Section 5.1. Crack initiation and growth due to IGSCC of the Class 1 stainless steel piping system is a detrimental aging effect for the large-bore piping. Therefore, the aging management program described in Section 4.3.2 will be continued during the extended operating period to manage this aging effect.

Table 2
Stainless Steel Large-bore Piping Components
Review of Plausible and Detrimental Aging Effects

Plausible Aging Effects	*Plausible Aging Mechanisms*	*Effect/Mech. Is Detrimental?*	*Remarks*
Crack Initiation and Growth	Thermal Fatigue	Yes	Includes small-bore piping > 1 inch Evaluated as TLAA in Section 7.2 Aging management program is described in Section 4.3.1
	IGSCC	Yes	Aging management program is described in Section 4.3.2

4.2.2 Stainless Steel Small-bore Piping and Valve Bodies

Commodity Description

The RRS contains instrument and drain piping which is 2 inch and smaller, and numerous small valves in the instrumentation and drain lines. This commodity group also includes the pipe fittings, thermowells and pressure taps of the flow venturis.

The Unit 1 small piping is part of the original plant and is Type TP304 stainless steel. It was designed and furnished to ANSI B31.7, Class 1, 1969 Edition. The valves range in size from ¾ inch to 2 inches. They are forged valves, constructed of Type T304 stainless steel. The body to bonnet closure is welded.

The Unit 2 small piping is part of the original plant and is Type TP304 and TP316 stainless steel. It was designed and furnished to ASME Section III, Class 1, 1971 Edition. The valves range in size from ¾ inch to 2 inches. They are forged valves, constructed of Type T304 stainless steel. The body to bonnet closure is welded. Any small-bore piping that is or has been replaced at Plant Hatch since 1984 has been type TP304L or TP316L, which are IGSCC resistant grades.

On both units, the pipe fittings and the pressure tap connections are made of Type F304 stainless steel and the thermowells are constructed of Type F316 stainless steel.

For both units, the internal environment is reactor water and the external environment is the drywell environment, which is benign. These environments are more completely described in Section 2.0 of this report.

Aging Effects

Plausible and Detrimental Aging Effects for Stainless Steel Small-bore Piping and Valve Bodies Components are listed in Table 3.

Crack initiation and growth due to thermal fatigue is described in Section 5.1. Small-bore (less than 1 inch) stainless steel piping and valve bodies are designed to withstand at least 7,000 full thermal cycles. The small-bore stainless steel components in the RRS will not experience 7,000 cycles during 60 years of operation. Therefore, crack initiation and growth due to thermal fatigue of small-bore stainless steel piping and valve bodies 1 inch and under is not a detrimental aging effect. Small-bore piping and valve body components 2 inches in diameter are included in the Stainless Steel Large-bore Piping commodity for evaluation of crack initiation and growth due to thermal fatigue.

Crack initiation and growth due to IGSCC is described in Section 5.1. Type 316 stainless steel, which some of the Unit 2 small-bore piping is made from, is resistant but not immune to crack initiation and growth due to IGSCC. The Unit 1 small-bore piping and some of the Unit 2 small-bore piping are made from Type 304 stainless steel. Crack initiation and growth due to IGSCC has not been observed in the small-bore welded joints or heat affected zones of Type 304 stainless steel at Plant Hatch. However, this is the same material and environment as the Plant Hatch large-bore piping welded joints that were found to have cracks attributed to IGSCC. Therefore, crack initiation and growth due to IGSCC will be treated as a detrimental aging effect for stainless steel small-bore piping and valve body components at Plant Hatch. The aging management program described in Section 4.3.2 will be continued, during the extended operating period, to manage this aging effect for the small-bore stainless steel piping and valve bodies.

Crack initiation and growth due to other causes is described in Section 5.1. The aging management program described in Section 4.3.3 will be continued, during the extended operating period, to manage crack initiation and growth for the small-bore stainless steel piping.

Loss of material due to crevice corrosion or pitting corrosion is described in Section 5.2. This aging effect is prevented from occurring in the small-bore stainless steel piping and valves at Plant Hatch by the aging management program described in Section 4.3.4. Because an aging management program is required to prevent this aging effect, it is considered a detrimental aging effect for the small-bore stainless steel piping and valves.

Table 3
Small-bore Stainless Steel Piping and Valves
Review of Plausible and Detrimental Aging Effects

Plausible Aging Effects	*Plausible Aging Mechanisms*	*Effect/Mech. Is Detrimental?*	*Remarks*
Crack Initiation and Growth	Thermal Fatigue	No	RRS Components will not exceed 7,000 cycles in 60 years
	IGSCC	Yes	**Aging management program is described in Section 4.3.2**
	Other	Yes	**Aging management program is described in Section 4.3.3**
Loss of Material	Crevice Corrosion Pitting Corrosion	Yes	Aging management program is described in Section 4.3.4

4.2.3 Pump Casings

Commodity Description

The recirculation pumps are variable speed, motor driven pumps, each rated at 45,200 gpm. The pump casing and cover form part of the RRS pressure boundary These components are constructed of statically-cast austenitic stainless steel (CASS): ASTM A351 Gr. CF8M (Unit 1) and ASME SA351 Gr. CF8M (Unit 2). The certified material test reports, submitted with these pumps, indicate that the maximum percentage of delta ferrite in any of these components is 14 percent. The Unit 1 reactor recirculation pumps were designed and furnished to the ASME Boiler and Pressure Vessel (B&PV) Code, Section III, 1968 Edition; and, the Unit 2 reactor recirculation pumps were designed and furnished to the ASME Standard Code for Pumps and Valves for Nuclear Power, 1968 draft Edition. For both units, the internal environment is reactor water and the external environment is the drywell environment, which is benign. These environments are more completely described in Section 2.0 of this report.

Aging Effects

Plausible and Detrimental Aging Effects for Pump Casings are listed in Table 4.

Crack initiation and growth due to IGSCC is described in Section 5.1. The Plant Hatch pump casing material is constructed from CASS. Crack initiation and growth due to IGSCC is not a detrimental aging effect for the pump casings satisfying the boundaries of Figure 4 in ASTM STP 756 (Ref. 7). All but one of these pumps at Hatch have a material content which satisfies the boundaries of Figure 4 of ASTM STP 756. One pump, 2B31-C001A, requires an aging management program due to its higher carbon content. The aging management program for this pump is discussed in Section 4.3.2.

Loss of fracture toughness due to thermal aging embrittlement of CASS is described in Section 5.2. Loss of fracture toughness due to thermal aging embrittlement of the pump casing is a detrimental aging effect at Plant Hatch. The aging management program for these pumps is discussed in Section 4.3.4.

Table 4
CASS Pump Casings
Review of Plausible and Detrimental Aging Effects

Plausible Aging Effects	***Plausible Aging Mechanisms***	***Effect/Mech. Is Detrimental?***	***Remarks***
Crack Initiation and Growth	IGSCC	Yes	Aging management program for 2B31-C001A is described in Section 4.3.2
Loss of Fracture Toughness	Thermal Aging Embrittlement	Yes	Aging management program is described in Section 4.3.5

4.2.4 Large-bore Valve Bodies

Commodity Description

The recirculation pump suction and discharge valves are 28" motor operated gate valves. The Unit 1 valves were designed and furnished to ASME B&PV Code Section III, 1968 edition; and the Unit 2 valves were designed and furnished to the ASME Standard Code for Pumps and Valves for Nuclear Power, 1968 draft Edition. The valve body and the bonnet form part of the recirculation system pressure boundary. These CASS components were statically-cast and are constructed of: ASTM A351 Gr. CF8M (Unit 1) and ASME SA-351 Gr. CF8M (Unit 2). Certified material test reports, submitted with these valves, indicate that the maximum percentage of delta ferrite in any of these components is 16 percent. For both units, the internal environment is reactor water and the external environment is the drywell environment, which is benign. These environments are more completely described in Section 2.0 of this report.

Aging Effects

Plausible and Detrimental Aging Effects for large-bore valve bodies are listed in Table 5.

Crack initiation and growth due to IGSCC is described in Section 5.1. The large-bore valve bodies are constructed from CASS. Crack initiation and growth due to IGSCC is not a detrimental aging effect for the valve bodies satisfying the boundaries of Figure 4 in ASTM STP 756 (Ref. 7). All but one of these valve bodies, have a material content which satisfies the boundaries of Figure 4 of ASTM STP 756. One valve body, 1B31-

F031A, requires an aging management program due to its higher carbon content. The aging management program for this valve body is discussed in Section 4.3.2 .

Loss of fracture toughness due to thermal aging embrittlement of CASS is described in Section 5.2. Loss of fracture toughness due of to thermal aging embrittlement the large-bore valve bodies is a detrimental aging effect at Plant Hatch. The aging management program for these valve bodies is discussed in Section 4.3.5.

Table No. 5
CASS Large-bore Valve Bodies
Review of Plausible and Detrimental Aging Effects

Plausible Aging Effects	*Plausible Aging Mechanisms*	*Effect/Mech. Is Detrimental?*	*Remarks*
Crack Initiation and Growth	IGSCC	Yes	Aging management program for 1B31-F031A is described in Section 4.3.2
Loss of Fracture Toughness	Thermal Aging Embrittlement	Yes	Aging management program is described in Section 4.3.5

4.2.5 Integral Attachments

Commodity Description

The RRS pressure boundary also contains integral attachments for the supports. The integral attachments for both units are made from plate material (type 304 stainless steel) and attached to the pressure boundary with a full penetration weld. They are not exposed to the process environment. Integral attachments (or lugs) are used for connecting component supports to the large-bore piping and pump casings. They are also used to support the component thermal insulation.

Aging Effects

There are no plausible aging effects for integral supports in non-aggressive environments. Therefore, there are no detrimental aging effects for the integral attachments. No further review of aging effects on the integral attachments is required for the extended operating period.

4.2.6 Pressure Boundary Bolting

Commodity Description

The pump casing to cover bolting and large-bore valve body to bonnet bolting is part of the pressure boundary. The bolts/studs and nuts are constructed of stainless steel material. The bolts/studs and nuts are not exposed to the reactor coolant environment. They are exposed to the drywell environment, which is benign.

Aging Effects

Plausible and Detrimental Aging Effects for Pressure Boundary Bolting are listed in Table 6.

Loss of preload to stress relaxation is described in Section 5.4. The high temperatures that Class 1 bolting is exposed to can result in a loss of preload. Loss of preload due to stress relaxation of the pressure boundary bolting is a detrimental aging effect at Plant Hatch. The aging management program for this aging effect is discussed in Section 4.3.6.

Table No. 6
Pressure Boundary Bolting
Review of Plausible and Detrimental Aging Effects

Plausible Aging Effects	*Plausible Aging Mechanisms*	*Effect/Mech. Is Detrimental?*	*Remarks*
Loss of Preload	Stress Relaxation	Yes	Aging management program is described in Section 4.3.6

4.3 Management of Detrimental Aging Effects

Section 4.2 identified the aging effects that require aging management. The affected commodity grouping and the detrimental aging effect(s) to be managed are listed below. The aging management programs that will be used to manage these effects are summarized in Table 7 and discussed in the following sections.

Table 7
Reactor Recirculation System
Aging Management Review Summary

Section No.	Aging Effect	Pressure Boundary Component Grouping	Aging Management Programs Required for License Renewal [a]
4.3.1	Crack Initiation and Growth due to Thermal Fatigue	Large-bore Stainless Steel Piping	▪ New Fatigue Monitoring Program for Class 1 piping ▪ ISI Program ▪ Technical Specification surveillance procedures
4.3.2	Crack Initiation and Growth due to IGSCC	Large-bore Stainless Steel Piping Small-bore Stainless Steel Piping and Valve Bodies Pump Casings (2B31-C001A Only) Large-bore Valve Bodies (1B31-F031A Only)	▪ Water Chemistry Procedures including HWC ▪ ISI Program (Specify that 2B31-C001A & 1B31-F031A be inspected) ▪ Technical Specification surveillance procedures
4.3.3	Crack Initiation and Growth due to Other Causes	Small-bore Stainless Steel Piping	▪ ISI Program ▪ Technical Specification surveillance procedures
4.3.4	Loss of Material due to Crevice Corrosion or Pitting Corrosion	Small-bore Stainless Steel Piping	▪ Water Chemistry Procedures ▪ Technical Specification surveillance procedures
4.3.5	Loss of Fracture Toughness due to Thermal Aging Embrittlement	Pump Casings Large-bore Valve Bodies	▪ ISI Program
4.3.6	Loss of Preload due to Stress Relaxation	Pressure Boundary Bolting	▪ ISI Program ▪ Technical Specification surveillance procedures

Note (a) The Maintenance Procedures, QA Program, and Operating Event Review Procedures apply to all aging effects and commodity groups.

4.3.1 Crack Initiation and Growth Due to Thermal Fatigue

The aging management program for crack initiation and growth due to thermal fatigue consists of the following activities:

- Fatigue Monitoring Program for Class 1 piping
- ISI Program
- Technical Specification Surveillance Procedures
- Maintenance Procedures
- QA Program
- Operating Event Review Procedures

Collectively, these activities provide the program to manage crack initiation and growth due to thermal fatigue for the RRS large-bore piping. See Section 5.1 for a discussion on thermal fatigue of RRS components. The attributes and features of this collection of activities that are important for managing this aging effect are described in the following paragraphs and in Table 8.

The Fatigue Monitoring Program (Section 6.1) will monitor the fatigue cumulative usage factor (CUF) for the bounding Class 1 piping locations 2 inches and larger. The formulas being developed for this program can be used to determine the fatigue CUF of the most critical piping location at any time or to project the 60-year fatigue CUF for that location.

The ISI program (Section 6.1) monitors the condition of RRS components, including the large-bore piping (Section 4.2.1). The ASME Code, Section XI contains multiple requirements for these components. They are subject to periodic volumetric and surface exams. They are also subject to a VT-2 inspection during the RRS pressure boundary testing. The ISI program specifies inspection frequencies and sample sizes for these examinations based on ASME Section XI requirements, regulations and plant commitments.

Technical Specification surveillance procedures (Section 6.1) provide guidance to monitor the RRS for system leakage. The plant QA program (Section 6.3) ensures that these activities are performed using procedures that have been reviewed and approved. The plant operating event review procedures (Section 6.3) provide guidance to assess plant and industry events. Corrective actions are initiated when appropriate. In some cases, maintenance procedures (Section 6.2) are used to implement corrective actions.

Demonstration

The TLAA evaluation in Section 7.2 shows that the design fatigue CUF, and the predicted CUF for the extended operating period is low for the RRS large-bore piping. The Fatigue Monitoring Program (Section 6.1) will monitor plant transients and track the fatigue CUF for the Class 1 piping.

The ISI Program, and Technical Specification surveillance procedures (Section 6.1) are linked to detection of cracking. Their scope includes the large-bore piping (Section 4.2.1).

Plant-specific and industry operating experience is reviewed as part of the ISI program and operating events review procedures (Section 6.3). The ongoing review of corrective actions and industry operating experience will provide the objective evidence to assure that the aging effects will be adequately managed.

The RRS is in good condition and performing its intended function. The above description links the aging management program to the detrimental aging effect. A review of plant operating history and inspections of the RRS demonstrate that the existing aging management activities credited have been, and will continue to be, effective in managing crack initiation and growth due to thermal fatigue (Section 5.1). In addition, a new aging management activity, as described in Section 6.1, will be established to monitor the fatigue CUF of the large-bore Class 1 piping. Therefore, there is reasonable assurance that detrimental aging effects will be adequately managed and the intended functions of the RRS will be maintained consistent with the CLB in the extended operating period.

Table 8
Aging Management Program Assessment
Crack Initiation and Growth due to Thermal Fatigue
in Stainless Steel Large-bore Piping

	Attributes	*Aging Management Program/Procedure*
1	Scope of the program includes the specific Structure, component or commodity (SCC) for the identified aging effect.	The Fatigue monitoring program, ISI program, and Technical Specification surveillance procedures include the RRS commodity grouping subject to the detrimental aging effect.
2	Preventive actions to mitigate or prevent aging degradation.	• The Fatigue monitoring program mitigates crack initiation and growth due to thermal fatigue by monitoring the events which contribute to the fatigue CUF. • The ISI program includes: ▪ Surface and volumetric inspections of welds in piping ▪ Pressure testing and VT-2 inspections. • Technical Specifications require leakage detection.
3	Parameters monitored or inspected are linked to the degradation of the particular SCC intended function.	The parameters monitored or inspected (plant transients, flaws, cracking, and leakage) are linked to maintaining pressure boundary and/or structural integrity, which ensures that RRS can perform its intended function.
4	The method of detection of the aging effects is described and performed in a timely manner.	The ISI program and Technical Specifications establish the methods of detection and the frequency of inspection, testing and surveillance based on ASME Section XI, regulations and plant commitments. The Fatigue monitoring program will specify the method of detection and will be performed at least once a year.
5	Monitoring and trending for timely corrective actions.	The ISI program provides for monitoring and trending. The Technical Specifications provide for monitoring. The Fatigue monitoring program will monitor and trend the events that can impact CUF.

	Attributes	*Aging Management Program/Procedure*
6	Acceptance criteria are included.	The ISI program, and Technical Specifications include acceptance criteria. The Fatigue monitoring program will include an acceptance criteria of CUF < 1.0.
7	Corrective actions, including root cause determination and prevention of recurrence, are included.	The ISI program, and Technical Specifications specify corrective actions. The Fatigue monitoring program will be governed by the plant QA program, which ensures corrective actions will be accomplished, including root cause determination and actions to prevent recurrence. The Fatigue Monitoring Program will also contain guidance that corrective actions can include revised calculations, flaw (crack) growth evaluations, increased ISI inspections, or replacement. Maintenance procedures may be used to perform corrective actions.
8	Confirmation process is included.	The ISI program, and Technical Specifications require reinspection following corrective actions. The plant QA program assures that corrective and preventive actions are accomplished and adequate.
9	Administrative controls should provide a formal review and approval process.	The plant QA program provides for the control of plant procedures and records associated with ISI, fatigue monitoring, and surveillance testing and inspections. These controls include a formal review and approval process.
10	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, are considered	The Fatigue monitoring program will be a new program so there is no operating experience for this program at this time. Plant-specific and industry operating experience is reviewed as part of the ISI program and Operating Event Report Review procedures.
11	Aging management programs and/or procedures are established by regulation and are subject to regulatory oversight.	The ISI program and Technical Specification surveillances are required by regulation and have been evaluated by the NRC, industry, and site organizations. The Fatigue monitoring program will be a new program at Plant Hatch. A similar program for the Hatch Reactor Pressure Vessel (RPV) is described in Section 4.2 of the Unit 1 FSAR and Section 5.4 of the Unit 2 FSAR. The RPV fatigue monitoring program has been evaluated by the NRC, industry, and site organizations.

4.3.2 Crack Initiation and Growth Due to IGSCC

The aging management program for crack initiation and growth due to IGSCC consists of the following activities:

- Water Chemistry Procedures including HWC
- ISI Program
- Technical Specification Surveillance Procedures
- Maintenance Procedures
- QA Program
- Operating Event Review Procedures

Collectively, these activities provide the program to manage crack initiation and growth due to IGSCC for the susceptible RRS commodities. See Section 5.1 for a discussion on IGSCC of RRS components. The attributes and features of this collection of activities that are important for managing this aging effect are described in the following paragraphs and in Table 9.

The cracking of stainless steel weldments due to IGSCC has been a significant issue for the boiling water reactor (BWR) primary coolant pressure boundary (PCPB) components. The technical review of this issue led to the development of procedures for repair, the use of more IGSCC resistant materials, augmented inspection programs for welds, and changes to, and monitoring of reactor coolant chemistry. Another major development, which can be used to investigate IGSCC, was the flaw detection analyses techniques. The ASME has played a major role on this issue and the current methods are documented in ASME, Section XI (Article IWA-3000, Mandatory Appendix III and Non-Mandatory Appendix A).

Cracking due to IGSCC has been the only significant aging effect actually experienced by the RRS components at Plant Hatch. Weld overlays (Section 6.2) were applied over identified cracks in the Unit 1 piping. The Unit 2 large-bore piping was replaced. The stresses in the RRS large-bore piping (Section 4.2.1) have been reduced by a number of design and installation practices including induction heating stress improvement (IHSI) and mechanical stress improvement process (MSIP) (Section 6.2). Most of the RRS components are now made from IGSCC resistant materials. One pump casing, 2B31-C001A, and one valve body, 1B31-F031A, require aging management for IGSCC because their material specifications are not within the boundaries of Figure 4 in ASTM 756, 1982 (Ref. 7).

The water chemistry procedures, including hydrogen water chemistry (HWC), provide guidelines to control the impurities in the reactor coolant to ensure that the RRS components are not subjected to a corrosive environment. See Section 6.2 for more details on the water chemistry procedures and HWC.

The ISI program (Section 6.1) monitors the condition of all the RRS components susceptible to IGSCC. NUREG-0313 (Ref. 4), Generic Letter 88-01(Ref. 9), and the ASME Code, Section XI contain multiple requirements for these components. Except for the small-bore piping, these commodities are subject to periodic volumetric and surface exams. All these commodities are subject to a VT-2 inspection during the RRS pressure boundary testing. If the pump or valve is disassembled for maintenance, the pump casing or valve body is subject to a VT-3 inspection. The ISI program specifies inspection frequencies and sample sizes for these examinations, based on the ASME Code, regulations, and plant commitments.

As described above, pump casing 2B31-C001A and valve body 1B31-F031A are more susceptible to IGSCC than the other pump casings and valve bodies. Therefore, the ISI program at Hatch will be enhanced to ensure that these locations are inspected. ASME Section XI, Category B-J requires inspection of 25 percent of the Class 1 piping welds over each 10-year Interval. The ISI program at Hatch will specify that at least one of the welds to these two components be inspected each interval during the extended operating period. Category B-L-2 and Category B-M-2 require only one pump casing and one valve body be inspected if more than one is disassembled during the interval. The Hatch ISI program will specify that each of these two components will be inspected each interval, during the extended operating period, if they are disassembled, even if another similar component has already been inspected.

Technical Specification surveillance procedures (Section 6.1) provide guidance to monitor the RRS for system leakage and water chemistry. The plant QA program (Section 6.3) ensures that these activities are performed using procedures that have been reviewed and approved. The plant operating event review procedures (Section 6.3) provide guidance to assess plant and industry events. Corrective actions are initiated when appropriate. In some cases, maintenance procedures (Section 6.2) are used to implement corrective actions.

Demonstration

Most RRS components are made of IGSCC resistant materials. Residual stresses in the large-bore piping welds have been minimized by IHSI and/or MSIP (section 6.2). Technical Specification surveillance procedures (Section 6.1) and the water chemistry procedures (Section 6.2) prevent a corrosive environment. The combination of these activities mitigates the possibility of cracking due to IGSCC (Section 5.1) for all of the RRS commodities. The use of HWC (Section 6.2) further mitigates the possibility of cracking due to IGSCC for the large-bore piping, valve bodies and pump casings. The three conditions that are needed for cracking due to IGSCC to occur are a susceptible material, an applied stress, and a corrosive environment. The material for most of the RRS commodities is not susceptible. The corrosive environment has been mitigated for all of the RRS commodities. All three conditions are being mitigated for the large-bore piping.

The ISI program (Section 6.1), and the Technical Specification surveillance procedures are linked to detection of cracking. Their scope includes the large-bore RRS piping (Section 4.2.1), the small-bore RRS piping (Section 4.2.2), the RRS pump casings (Section 4.2.3), and the large-bore RRS valve bodies (Section 4.2.4).

Plant-specific and industry operating experience is reviewed as part of the ISI program and operating event review procedures (Section 6.3). The ongoing review of corrective actions and industry operating experience will provide the objective evidence to assure that the aging effects will be adequately managed.

The RRS is in good condition and performing its intended function. Most of the RRS components are made from IGSCC resistant materials. The above description links the aging management program to the detrimental aging effect. A review of plant operating history and inspections of the RRS demonstrate that the existing aging management activities credited have been, and should continue to be, effective in managing crack initiation and growth due to IGSCC. Crack initiation and growth has been identified, monitored, and corrected when required. Therefore, there is reasonable assurance that this detrimental aging effect will be adequately managed and the intended functions of the RRS will be maintained consistent with the CLB in the extended operating period.

Table 9
Aging Management Program Assessment
Crack Initiation and Growth due to IGSCC
in Stainless Steel Piping, Valve Bodies, and Pump Casings

	Attributes	*Aging Management Program/Procedure*
1	Scope of the program includes the specific Structure, component or commodity (SCC) for the identified aging effect.	The ISI program, Technical Specification surveillance procedures and water chemistry procedures include the RRS commodity groupings subject to this detrimental aging effect. The ISI Program includes augmented inspections required by regulation or plant commitments.
2	Preventive actions to mitigate or prevent aging degradation.	• Corrosion resistant materials for all RRS commodities; IGSCC resistant materials and IHSI for large-bore piping • Process environment is controlled by water chemistry procedures and HWC. • The ISI program includes: ▪ surface and volumetric inspections of welds in piping and valve bodies > 4" in diameter and pump casings ▪ pressure testing and VT-2 inspection ▪ VT-3 inspection of pump casing and valve body if pump or valve is disassembled for maintenance. • Technical Specifications require leakage detection and water chemistry limits.
3	Parameters monitored or inspected are linked to the degradation of the particular SCC intended function.	The parameters monitored or inspected (cracking, flaws, leakage and water chemistry) are linked to maintaining pressure boundary and/or structural integrity, which ensures that RRS can perform its intended function.
4	The method of detection of the aging effects is described and performed in a timely manner.	The ISI program and Technical Specifications establish the methods of detection and the frequency of inspection, testing and surveillance based on ASME Section XI, regulations and plant commitments. Water chemistry is monitored daily and methods of detection are specified.
5	Monitoring and trending for timely corrective actions.	The ISI program and the water chemistry procedures provide for monitoring and trending. The Technical Specifications provide for monitoring.

	Attributes	*Aging Management Program/Procedure*
6	Acceptance criteria are included.	The ISI program, Technical Specifications and the water chemistry procedures all include acceptance criteria.
7	Corrective actions, including root cause determination and prevention of recurrence, are included.	The ISI program, and Technical Specifications, and the water chemistry procedures specify corrective actions. The plant QA program ensures corrective actions will be accomplished, including root cause determination and actions to prevent recurrence.
8	Confirmation process is included.	The ISI program, and Technical Specifications require reinspection following corrective actions. The plant QA program assures that corrective and preventive actions are accomplished and adequate.
9	Administrative controls should provide a formal review and approval process.	The plant QA program provides for the control of plant procedures and records associated with ISI, water chemistry, and surveillance testing and inspections. These controls include a formal review and approval process.
10	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, are considered	Plant-specific and industry operating experience is reviewed as part of the ISI program and Operating Event Report Review procedures.
11	Aging management programs and/or procedures are established by regulation and are subject to regulatory oversight.	The ISI program and Technical Specification surveillances are required by regulation. The water chemistry procedures implement industry recommendations. All of these programs have been evaluated by the NRC, industry, and site organizations.

4.3.3 Crack Initiation and Growth Due to Other Causes

The aging management program for crack initiation and growth due to other causes consists of the following activities:

- ISI Program
- Technical Specification Surveillance Procedures
- Maintenance Procedures
- QA Program
- Operating Event Review Procedures

Collectively, these activities provide the program to manage crack initiation and growth due to other causes for the small-bore piping. See Section 5.1 for a discussion on other causes of crack initiation and growth in small-bore piping (Section 4.2.2). The attributes and features of this collection of activities that are important for managing this aging effect are described in the following paragraphs and in Table 10.

Maintenance procedures (Section 6.2) provide for monitoring and trending vibration on major pumps and motors. Identifying excessive vibration and correcting it can prevent vibrational fatigue.

Operating event review procedures (Section 6.3) will also identify concerns about other causes of cracking and appropriate actions will be taken.

Under the ISI program (Section 6.1) the small-bore piping is subject to a VT-2 inspection during the RRS pressure boundary testing per the requirements of Category B-P of ASME Section XI. Volumetric and surface examinations of small-bore piping under 2 ½ inches is not effective.

Technical Specification surveillance procedures (Section 6.1) provide guidance to monitor the RRS for system leakage. Very small leak rates are detectable. If there is pressure boundary leakage, per Technical Specification 3.4.4.C, the unit is shutdown. The leak rate from RRS small-bore piping cannot exceed the system makeup capacity and any leakage is contained and processed by the radwaste system. The plant QA program (Section 6.3) ensures that these activities are performed using procedures that have been reviewed and approved. The plant operating event review procedures (Section 6.3) provide guidance to assess plant and industry events. Corrective actions are initiated when appropriate.

Demonstration

Preventive actions to mitigate this aging effect are accomplished by design coupled with the maintenance procedures (Section 6.2) and the operating event review procedures.

The ISI program, and Technical Specification surveillance procedures (Section 6.1) are linked to detection of cracking. Their scope includes the small-bore RRS piping (Section 4.2.2).

There has been one event at Plant Hatch involving the RRS small-bore piping. This event occurred in 1987 at the socket weld for a line installed in 1984. There has been no recurrence. Plant-specific and industry operating experience is reviewed as part of the ISI program and operating event review procedures (Section 6.3). The ongoing review of corrective actions and industry operating experience will provide the objective evidence to assure that the aging effects will be adequately managed.

The RRS is in good condition and performing its intended function. The above description links the aging management program to the detrimental aging effect. A review of plant operating history and inspections of the RRS demonstrate that the existing aging management activities credited have been, and should continue to be, effective in managing crack initiation and growth due to other causes (Section 5.1). Crack initiation and growth in small-bore stainless steel piping has been identified and corrected without loss of intended function. Therefore, there is reasonable assurance that this detrimental aging effect will be adequately managed and the intended functions of the RRS will be maintained consistent with the CLB in the extended operating period.

Table 10
Aging Management Program Assessment
Crack Initiation and Growth Due to Other Causes
in Small-bore Stainless Steel Piping

	Attributes	*Aging Management Program/Procedure*
1	Scope of the program includes the specific Structure, component or commodity (SCC) for the identified aging effect.	The ISI program, Technical Specification surveillance procedures and maintenance procedures includes the RRS commodity grouping subject to this detrimental aging effect.
2	Preventive actions to mitigate or prevent aging degradation.	• Maintenance procedures monitor vibration of pumps & motors. • ISI program includes pressure testing and VT-2 inspection. • Technical Specifications require leakage detection.
3	Parameters monitored or inspected are linked to the degradation of the particular SCC intended function.	The parameters monitored or inspected (leakage and vibration) are linked to maintaining pressure boundary and/or structural integrity, which ensures that RRS can perform its intended function.
4	The method of detection of the aging effects is described and performed in a timely manner.	The ISI program and Technical Specifications establish the methods of detection and the frequency of inspection, testing and surveillance based on ASME Section XI, regulations and plant commitments. The maintenance procedures specify frequency and methods of detection.
5	Monitoring and trending for timely corrective actions.	The ISI program and maintenance procedures provide for monitoring and trending. The Technical Specifications provide for monitoring.
6	Acceptance criteria are included.	The ISI program, Technical Specifications and the maintenance procedures all include acceptance criteria.
7	Corrective actions, including root cause determination and prevention of recurrence, are included.	The ISI program, maintenance procedures, and Technical Specifications specify corrective actions. The plant QA program ensures corrective actions will be accomplished, including root cause determination and actions to prevent recurrence.
8	Confirmation process is included.	The ISI program, and Technical Specifications require reinspection following corrective actions. The plant QA program assures that corrective and preventive actions are accomplished and adequate.
9	Administrative controls should provide a formal review and approval process.	The plant QA program provides for the control of plant procedures and records associated with ISI, maintenance procedures, and surveillance testing and inspections. These controls include a formal review and approval process.
10	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, are considered	Plant-specific and industry operating experience is reviewed as part of the ISI program and Operating Event Report Review procedures.
11	Aging management programs and/or procedures are established by regulation and are subject to regulatory oversight.	The ISI program and Technical Specification surveillances are required by regulation. Maintenance procedures implement industry guidance. These programs have been evaluated by the NRC, industry, and site organizations.

4.3.4 Loss of Material Due to Crevice Corrosion or Pitting Corrosion

The aging management program for loss of material due to crevice corrosion or pitting corrosion consists of the following activities:

- Water Chemistry Procedures
- Technical Specification Surveillance Procedures
- Maintenance Procedures
- QA Program
- Operating Event Review Procedures

Collectively, these activities provide the program to manage loss of material due to crevice corrosion or pitting corrosion for the small-bore piping and valves. See Section 5.2 for a discussion on crevice corrosion and pitting corrosion of RRS components. The attributes and features of this collection of activities that are important for managing this aging effect are described in the following paragraphs and in Table 11.

The water chemistry procedures provide guidelines to control the impurities in the reactor coolant to ensure that the RRS components are not subjected to a corrosive environment. See Section 6.2 for more details on the water chemistry procedures.

Technical Specification surveillance procedures (Section 6.1) provide guidance to monitor the RRS water chemistry. The plant QA program (Section 6.3) ensures that these activities are performed using procedures that have been reviewed and approved. The plant operating event review procedures (Section 6.3) provide guidance to assess plant and industry events. Corrective actions are initiated when appropriate. In some cases, maintenance procedures (Section 6.2) are used to implement corrective actions.

Demonstration

The water chemistry procedures (Section 6.1) prevent loss of material due to crevice corrosion or pitting corrosion (Section 5.2) for the small-bore stainless steel piping and valves.

The Technical Specification surveillance procedures (Section 6.1) are linked to mitigating corrosion. Their scope includes the small-bore RRS piping (Section 4.2.2).

Plant-specific and industry operating experience is reviewed as part of the operating event review procedures (Section 6.3). The ongoing review of corrective actions and industry operating experience will provide the objective evidence to assure that the aging effects will be adequately managed.

The RRS is in good condition and performing its intended function. The above description links the aging management program to the detrimental aging effect. A review of plant operating history and inspections of the RRS demonstrate that the existing aging management activities credited have been, and should continue to be, effective in managing loss of material due to crevice corrosion or pitting corrosion. This aging effect has not been found in these components. Therefore, there is reasonable assurance that this detrimental aging effect will be adequately managed and the intended functions of the RRS will be maintained consistent with the CLB in the extended operating period.

Table 11
Aging Management Program Assessment
Loss of Material due to Crevice Corrosion or Pitting Corrosion
in Small-bore Stainless Steel Piping

	Attributes	*Aging Management Program/Procedure*
1	Scope of the program includes the specific Structure, component or commodity (SCC) for the identified aging effect.	The Technical Specification surveillance procedures and water chemistry procedures include the RRS commodity grouping subject to this detrimental aging effect.
2	Preventive actions to mitigate or prevent aging degradation.	• Corrosion resistant materials for all RRS commodities • Process environment is controlled by water chemistry procedures. • Technical Specifications require water chemistry limits.
3	Parameters monitored or inspected are linked to the degradation of the particular SCC intended function.	The parameters monitored (water chemistry) are linked to maintaining pressure boundary and/or structural integrity, which ensures that RRS can perform its intended function.
4	The method of detection of the aging effects is described and performed in a timely manner.	The Technical Specifications establish the methods of detection and the frequency of surveillance based on regulations. Water chemistry is monitored daily and methods of detection are specified.
5	Monitoring and trending for timely corrective actions.	The water chemistry procedures provide for monitoring and trending. The Technical Specifications provide for monitoring.
6	Acceptance criteria are included.	The Technical Specifications and the water chemistry procedures include acceptance criteria.
7	Corrective actions, including root cause determination and prevention of recurrence, are included.	The Technical Specifications and the water chemistry procedures specify corrective actions. The plant QA program ensures corrective actions will be accomplished, including root cause determination and actions to prevent recurrence.
8	Confirmation process is included.	The Technical Specifications require reinspection following corrective actions. The plant QA program assures that corrective and preventive actions are accomplished and adequate.
9	Administrative controls should provide a formal review and approval process.	The plant QA program provides for the control of plant procedures and records associated with water chemistry, and surveillance testing. These controls include a formal review and approval process.
10	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, are considered	Plant-specific and industry operating experience is reviewed as part of the Operating Event Report Review Procedures.
11	Aging management programs and/or procedures are established by regulation and are subject to regulatory oversight.	The Technical Specification surveillances are required by regulation. The water chemistry procedures implement industry recommendations. These programs have been evaluated by the NRC, industry, and site organizations.

4.3.5 Loss of Fracture Toughness Due to Thermal Aging Embrittlement

The aging management program for loss of fracture toughness due to thermal aging embrittlement consists of the following activities:

- ISI Program
- Maintenance Procedures
- QA Program
- Operating Event Review Procedures

Collectively, these activities provide the program to manage loss of fracture toughness due to thermal aging embrittlement for the susceptible RRS commodities. See Section 5.3 for a discussion on thermal aging embrittlement of RRS components. The attributes and features of this collection of activities that are important for managing this aging effect are described in the following paragraphs and in Table 12.

Loss of fracture toughness due to thermal aging embrittlement is discovered during plant operations by the existence of cracks. The ISI program (Section 6.1) monitors the condition of all the RRS components susceptible to loss of fracture toughness. The ASME Code, Section XI contains multiple requirements for these components. These commodities are subject to periodic volumetric and surface exams. If a pump or valve is disassembled for maintenance, the pump casing or valve body is subject to a VT-3 inspection. Relevant indications found during the VT-3 examination are subject to supplemental examinations, in order to characterize the indications further, and potential engineering evaluations. If flaws detected and sized by those supplemental examinations exceed acceptance criteria, the flaw evaluation procedures specified in IWB-3640 will be used. The NRC safety evaluation for BAW-2243A (Ref. 18) approved these procedures specifically for CASS valve bodies; however, they may be applied to other CASS components, such as the pump casings, as discussed in EPRI-TR-106092 (Ref. 8). The ASME Code specifies inspection frequencies, sample sizes, etc. for these examinations.

The plant QA program (Section 6.3) ensures that these activities are performed using procedures that have been reviewed and approved. The plant operating event review procedures (Section 6.3) provide guidance to assess plant and industry events. Corrective actions are initiated when appropriate. In some cases, maintenance procedures (Section 6.2) are used to implement corrective actions.

Demonstration

The scope of the ISI program (Section 6.1) includes the RRS pump casings (Section 4.2.3) and the large-bore RRS valve bodies (Section 4.2.4).

There has been no experience of this aging effect at Plant Hatch. Plant-specific and industry operating experience is reviewed as part of the ISI program and operating event review procedures (Section 6.3). The ongoing review of corrective actions and industry operating experience will provide the objective evidence to assure that the aging effects will be adequately managed.

The RRS is in good condition and performing its intended function. The above description links the aging management program to the detrimental aging effect. A review of plant operating history and inspections of the RRS demonstrate that the existing aging management activities credited have been, and should continue to be, effective in managing loss of fracture toughness due to thermal aging embrittlement (Section 5.2). Therefore, there is reasonable assurance that this detrimental aging effect will be adequately managed and the intended functions of the RRS will be maintained consistent with the CLB in the extended operating period.

Table 12
Aging Management Program Assessment
Loss of Fracture Toughness due to Thermal Aging Embrittlement of Stainless Steel Large-bore Valve Bodies and Pump Casings

	Attributes	*Aging Management Program/Procedure*
1	Scope of the program includes the specific Structure, component or commodity (SCC) for the identified aging effect.	The ISI program includes the RRS commodity groupings subject to this detrimental aging effect.
2	Preventive actions to mitigate or prevent aging degradation.	The ISI program includes: • Surface and volumetric inspections of welds in valve bodies and pump casings • VT-3 inspection of valve bodies and pump casing • Flaw evaluation per IWB-3640
3	Parameters monitored or inspected are linked to the degradation of the particular SCC intended function.	The parameters inspected (cracking and flaws) are linked to maintaining pressure boundary and/or structural integrity, which ensures that RRS can perform its intended function.
4	The method of detection of the aging effects is described and performed in a timely manner.	The ISI Program establishes the methods of detection and the frequency of inspection based on ASME Section XI, regulations and plant commitments.
5	Monitoring and trending for timely corrective actions.	The ISI Program provides for monitoring and trending.
6	Acceptance criteria are included.	The ISI Program includes acceptance criteria.
7	Corrective actions, including root cause determination and prevention of recurrence, are included.	The ISI Program specifies corrective actions. The plant QA program ensures corrective action will be accomplished, including root cause determination and actions to prevent recurrence.
8	Confirmation process is included.	The ISI Program requires reinspection following corrective actions. The plant QA program assures that corrective and preventive actions are accomplished and adequate.
9	Administrative controls should provide a formal review and approval process.	The plant QA program provides for the control of plant procedures and records associated with ISI inspections. These controls include a formal review and approval process.
10	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, are considered	Plant-specific and industry operating experience is reviewed as part of the ISI program and Operating Event Report Review Procedures.
11	Aging management programs and/or procedures are established by regulation and are subject to regulatory oversight.	The ISI program is required by regulation. This program has been evaluated by the NRC, industry, and site organizations.

4.3.6 Loss of Preload Due to Stress Relaxation

The aging management program for loss of preload due to stress relaxation consists of the following activities:

- ISI Program
- Technical Specification Surveillance Procedures
- Maintenance Procedures
- QA Program
- Operating Event Review Procedures

Collectively, these activities provide the program to manage loss of preload due to stress relaxation for the pressure boundary bolting. See Section 5.4 for a discussion on stress relaxation of pressure boundary bolting. The attributes and features of this collection of activities that are important for managing this aging effect are described in the following paragraphs and Table 13.

Maintenance procedures (Section 6.2) provide instructions to ensure proper preload is applied. This cannot prevent loss of preload, but it does mitigate the aging effect. If leakage is found, maintenance procedures would then be used for corrective actions.

Under the ISI program (Section 6.1) the pressure boundary bolting is subject to a VT-2 inspection during the RRS pressure boundary testing per the requirements of Category B-P of ASME Section XI. This inspection is performed near the end of outages to ensure that all unacceptable leakage is found and corrected prior to returning the unit to operation. A review of plant experience found that this inspection has previously found leakage attributed to loss of preload.

Technical Specification surveillance procedures (Section 6.1) provide guidance to monitor the RRS for system leakage. If there is pressure boundary leakage, per Technical Specification 3.4.4.C, the unit is shutdown. The plant QA program (Section 6.3) ensures that these activities are performed using procedures that have been reviewed and approved. The plant operating event review procedures (Section 6.3) provide guidance to assess plant and industry events. Corrective actions are initiated when appropriate.

Demonstration

Preventive actions to mitigate this aging effect are accomplished by proper torquing with the maintenance procedures (Section 6.2).

The ISI program, and Technical Specification surveillance procedures (Section 6.1) are linked to detection of leakage. Their scope includes the pressure boundary bolting (Section 4.2.6).

The RRS is in good condition and performing its intended function. The above description links the aging management program to the detrimental aging effect. A review of plant operating history and inspections of the RRS demonstrate that the existing aging management activities credited have been, and should continue to be, effective in managing loss of preload due to stress relaxation (Section 5.4). Loss of preload has been identified and corrected without loss of the system's intended function. Therefore, there is reasonable assurance that this detrimental aging effect will be adequately managed and the intended functions of the RRS will be maintained consistent with the CLB in the extended operating period.

Table 13
Aging Management Program Assessment
Loss of Preload due to Stress Relaxation of Pressure Boundary Bolting

	Attributes	*Aging Management Program/Procedure*
1	Scope of the program includes the specific Structure, component or commodity (SCC) for the identified aging effect.	The ISI program, Technical Specification surveillance procedures and maintenance procedures include the RRS commodity grouping subject to this detrimental aging effect.
2	Preventive actions to mitigate or prevent aging degradation.	• Maintenance procedures ensure that bolts are properly torqued. • The ISI program includes pressure testing with a VT-2 inspection. • Technical Specifications require leakage detection.
3	Parameters monitored or inspected are linked to the degradation of the particular SCC intended function.	The parameters monitored or inspected (torque and leakage) are linked to maintaining pressure boundary and/or structural integrity, which ensures that RRS can perform its intended function.
4	The method of detection of the aging effects is described and performed in a timely manner.	The ISI program and Technical Specifications establish the methods of detection and the frequency of inspection, testing and surveillance based on ASME Section XI, regulations and plant commitments.
5	Monitoring and trending for timely corrective actions.	The ISI program provides for monitoring and trending. The Technical Specifications provide for monitoring.
6	Acceptance criteria are included.	The ISI program, Technical Specifications and the maintenance procedures all include acceptance criteria.
7	Corrective actions, including root cause determination and prevention of recurrence, are included.	The ISI program and Technical Specifications specify corrective actions. The plant QA program ensures corrective actions will be accomplished, including root cause determination and actions to prevent recurrence. The maintenance procedures may be used to perform corrective actions.
8	Confirmation process is included.	The ISI program, and Technical Specifications require reinspection following corrective actions. The plant QA program assures that corrective and preventive actions are accomplished and adequate.
9	Administrative controls should provide a formal review and approval process.	The plant QA program provides for the control of plant procedures and records associated with ISI, maintenance procedures, and surveillance testing and inspections. These controls include a formal review and approval process.
10	Operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs, are considered	Plant-specific and industry operating experience is reviewed as part of the ISI program and Operating Event Report Review procedures.
11	Aging management programs and/or procedures are established by regulation and are subject to regulatory oversight.	The ISI program and Technical Specification surveillances are required by regulation. Maintenance procedures implement industry guidance. These programs have been evaluated by the NRC, industry, and site organizations.

5.0 AGING EFFECTS

This Section provides a discussion of the various aging effects evaluated in Section 4.0 of this report. In most cases this discussion includes identification of the mechanisms that cause the effect, material/environment combinations that are particularly susceptible and particularly resistant to the mechanism, and identification of typical methods used in the industry to prevent, mitigate, and monitor the effect.

5.1 Cracking

Cracking in Stainless Steel can be caused by several mechanisms. The mechanisms that are plausible for the RRS are:

- Thermal Fatigue
- Stress Corrosion Cracking (IGSCC)
- Other Causes

Thermal Fatigue

Fatigue is a general term that describes the structural deterioration that can occur as the result of repeated stress/strain cycles caused by fluctuating loads or temperatures. After repeated cyclic loading, if sufficient localized microstructural damage has been accumulated, crack initiation can occur at the most highly affected locations. Subsequent cyclic loading and/or thermal stress can cause crack growth.

The presence of an oxidative environment, as the Hatch pressure boundary piping system was prior to using HWC (Section 6.2), can accelerate the fatigue crack initiation and propagation process. This is commonly referred to as environmentally assisted fatigue. The NRC questioned the adequacy of the ASME Code fatigue curves to account for this effect. However, testing performed by the industry (Ref. 11, 12, and 13) has shown that the combination of existing conservatisms is such that explicit treatment of reactor water environmental effects in fatigue design analysis is not considered necessary.

The Welding Research Council (Ref. 16) studied piping failures in domestic nuclear power plants through 1991. If failure incidents (major breaks or splits) are separated from less critical crack and leakage observation incidents, the Welding Research Council identified no failures for piping classified as ASME B&PV Code, Section III, Class I that were greater than or equal to 2 inches in diameter. There has been no cracking due to thermal fatigue on any of these components at Plant Hatch.

Cracking due to fatigue is a plausible aging effect for the Class 1 piping, valves bodies, and pump casings. At Plant Hatch, the design against fatigue cracking in the large-bore stainless steel piping (Section 4.2.1) is based on satisfaction of cumulative fatigue usage factor limits using the fatigue curves from ASME B&PV Codes. The Unit 1 RRS piping fatigue limits are based on ASME Section III, 1983 Edition with Addenda to and including Winter of 1984; and, the Unit 2 limits are based on 1980 Edition with addenda to and including Winter of 1981.

The 2 inch Class 1 small-bore piping is included in the design fatigue calculations performed for Class 1 large-bore stainless steel piping. Crack initiation and growth due to fatigue is a detrimental aging effect at Plant Hatch for the large-bore stainless steel piping. The aging management program for this aging effect is described in Section 4.3.1. Fatigue of Class 1 large-bore stainless steel piping is a TLAA and is discussed in Section 7.2.

At Plant Hatch, the design against fatigue cracking for Class 1 small-bore stainless steel piping and valve bodies (Section 4.2.2) 1 inch in diameter and smaller is based on the ASME Section III, Class 1, 1971 Edition cyclic load evaluations. The inherent flexibility of the small-bore piping reduces the loading due to thermal cycles. These components, and Class 2 and 3 components, are designed for at least 7,000 full-temperature cycles. No aging management program is needed to manage fatigue of these components because the RRS system will not exceed 7,000 cycles during 60 years.

At Plant Hatch, the RRS Pump casing (Section 4.2.3) design considered the potential for thermal fatigue cycling to occur. Actual fatigue usage values have not been determined for these components at Hatch. Industry evaluations (Ref. 5) have shown that typically the recirculation pump casing fatigue usage is very low. These evaluations determined that the fatigue usage factors are insignificant in accordance with the code criteria used in the design of these pumps. The pump casings at Plant Hatch have similar design features to those used in the industry study, including wall thickness, material and cyclic loading. These results demonstrate that the design of the Plant Hatch pumps will remain valid, with respect to cumulative fatigue usage, for the period of extended operation. No further review of this aging effect is required for the extended operating period.

Per GE SIL 459, Supplement 1 (Ref. 14) thermal fatigue cracking of pump covers due to localized thermal effects (mixing of colder seal purge flow with reactor coolant) has been observed in several Byron Jackson 1st and 2nd generation recirculation pumps during ISI inspections. One instance of this was believed to result in pressure boundary leakage, but testing failed to confirm this. The pump in this case was altered to allow leakage to be monitored, collected, and measured. No detectable leakage was identified. Plant Hatch has Byron Jackson 2nd generation pumps, but the covers will be replaced with the new Byron Jackson 4th generation design. The parts have been purchased and implementation is currently planned for 2001 on Unit 2 and 2002 on Unit 1. The new design features a redesigned pump shaft and cover , along with a new heater and cooler. Based on factory testing and operation at two power plants, this design eliminates the formation of cracks. Dye penetrant inspections on pumps featuring the 4th generation design revealed no cracks after two full years of operation. Since the pump covers will be replaced, no aging management program is required.

At Plant Hatch, the design of RRS large-bore valve bodies (Section 4.2.4) considered the potential for thermal fatigue cycling to occur. Actual fatigue usage values have not been determined for these components at Hatch. Industry evaluations (Ref. 5) have shown that typically the fatigue usage is less than 0.25 for 40 years, which translates to less than 0.4 for 60 years. Plant Hatch large-bore valve bodies have similar design features to those used in the industry study, including wall thickness, material and cyclic loading. The BWR Primary Coolant Pressure Boundary License Renewal Industry Report (Ref. 5) states, "For fatigue to be resolved, the component must have no experience of flaws induced by fatigue and either typical evaluations of the component with similar design and operation or plant-specific evaluations show that the fatigue usage factor is less than 0.4 for SS or 0.25 for CS for at least 40 years." Therefore, it is reasonable to conclude that components with a life of plant projected fatigue usage less than 0.4 do not need to be included in an aging management program. The design of the Plant Hatch large-bore valve bodies will remain valid, with respect to cumulative fatigue usage, for the period of extended operation. No further review of this aging effect is required for the extended operating period.

Stress Corrosion Cracking

Stress corrosion cracking occurs through the combination of high stress, a corrosive environment, and a susceptible material. Stress corrosion cracking can be categorized as:

- Intergranular Stress Corrosion Cracking (IGSCC)
- Transgranular Stress Corrosion Cracking (TGSCC) *
- Irradiation Assisted Stress Corrosion Cracking (IASCC) *
    - * Not discussed in this report

Intergranular stress corrosion cracking (IGSCC)

IGSCC refers to cracking which progresses along the microstructural grain boundaries. Three conditions must be present simultaneously to produce IGSCC: a susceptible material, a tensile stress (applied and/or residual), and a corrosive environment. The elimination of any one of these three factors or the reduction of one of these factors below threshold levels significantly reduces the potential for IGSCC.

IGSCC is typically associated with materials containing excessive grain boundary precipitation or impurity segregation. Although IGSCC usually occurs in fluid mediums with high dissolved oxygen (> 100 ppb), it can also occur in a low oxygen environment. Preferential grain boundary precipitation of carbides in austenitic stainless steels and nickel-base alloys leads to a localized depletion of chromium in the vicinity of the grain boundary. This process is known as sensitization and renders the material susceptible to IGSCC.

Intergranular attack (IGA), also known as intergranular corrosion, is similar in some respects to IGSCC; however, it is distinguished from IGSCC in that stress is not necessary for it to proceed. Generally, materials and conditions that are susceptible to IGSCC will also be susceptible to IGA. For purposes of this report, IGSCC will include IGA.

Since the 1960s, the susceptibility of BWR components to IGSCC has been known. Significant work, research, and testing have been done to understand the mechanisms and develop water chemistries and alternatives for the mitigation of IGSCC of reactor internals. The latest guidelines and technical bases are found in the EPRI BWR Water Chemistry Guidelines - 1996 Revision (Ref. 19). This document strengthens the recommendation for HWC (Section 6.2), provides methodology for plant specific water chemistry program development, and discusses the side effects of HWC.

Conductivity is a measure of impurities in the water. Studies have shown that certain impurities such as sulfate and chloride accelerate initiation of IGSCC and promote high crack growth rates. BWR plant measured conductivities have improved significantly over the years. Even the purest water will not provide immunity to IGSCC in the BWR, but good water will delay the initiation.

The Unit 1 large-bore piping (Section 4.2.1) is made of stainless steel and was modified by applying IHSI and/or MSIP to each circumferential weld joint (Section 6.2). In addition, IGSCC-resistant weld overlays (Section 6.2) were applied to the weldments with crack indications. There are 52 Category E welds at Plant Hatch, Unit 1. Category E welds are those with known flaws that have been reinforced by an acceptable weld overlay or have been mitigated by an IHSI treatment with subsequent qualified examination to verify the extent of cracking. Inspections of the original welds and weld overlays subsequent to application of the overlays have determined that no new or existing cracks have propagated into the weld overlay. Therefore, there has been no significant cracking found in this piping. The IHSI process was applied to the majority of the remaining welds within the non-isolable portion of the Unit 1 Reactor Coolant pressure boundary within the scope of the NUREG-0313 (Ref. 4). Hydrogen water chemistry (HWC) was also adopted to reduce the oxygen and peroxide content and thereby mitigate the corrosive environment (Section 6.2).

The Unit 2 large-bore piping was replaced with 316NG stainless steel, which is an IGSCC-resistant material. The welding material for the pipe weldments was limited to Type 308L except for dissimilar metal welds where Type 309L was used. Design, fabrication, and installation practices were used during the replacement to reduce the stress levels in the material. Applying IHSI and/or MSIP (Section 6.2) to each circumferential weld joint was one of these practices. Other practices included minimizing geometric discontinuities and establishing heat treatment, hot forming, and welding procedures that incorporated the results of industry research. HWC (Section 6.2) was also adopted to reduce the oxygen and peroxide content and thereby mitigate the corrosive environment. There has been no cracking due to IGSCC, that exceeded acceptance criteria, found in this piping since its replacement. Therefore, there has been no significant cracking in this piping.

The small-bore piping (Section 4.2.2) is constructed from stainless steel that is resistant but not immune to crack initiation and growth due to IGSCC.

ASTM 756, 1982 (Ref. 7) provides a reasonable basis for the determination of the susceptibility of CASS to IGSCC as a function of carbon and ferrite levels. Cast materials with low carbon content and greater than 5 percent ferrite content are considered resistant to sensitization. Most large bore valve bodies (Section 4.2.3) and pump casings (Section 4.2.4) are made of CASS material that is within the boundaries of Figure 4 in ASTM 756. Therefore, crack initiation and growth due to IGSCC is not a detrimental aging effect for these components.

At Plant Hatch, there has been no experience of significant cracking induced by IGSCC on either unit since these repairs and improvements were made. In all cases, at least two, and in some cases all three, of the conditions required to cause IGSCC have been successfully mitigated. The aging management program described in Section 4.3.2 will be continued during the extended operating period to manage this aging effect for the RRS components.

Other Causes

The principal aging effect for small-bore piping is potential cracking at the welded joint or in the base metal at or near weld heat-affected zones. Such cracking may be caused by growth of fabrication defects under normal service loads, or by initiation and growth of service-induced flaws caused by one or more aging mechanisms, such as thermal or high cycle fatigue. Since cracking due to thermal fatigue and IGSCC have already been discussed, this section pertains to cracking due to any other mechanism.

One such mechanism is high cycle fatigue. High cycle fatigue in small diameter piping components has been observed but design review and necessary modifications have essentially eliminated concerns (Ref. 17). Most of these failures have been attributed to design or installation problems. As such, these failures are normally detected early in plant life and action is taken to correct them.

At Plant Hatch, the small-bore piping (Section 4.2.2) design accounts for all known loads and stresses, including vibration. The larger the pipe, the less susceptible it is to vibrational (or high cycle) fatigue.

A review of Plant Hatch operating performance reveals a small number of small-bore piping failures. Most of these failures are in 1 inch or smaller pipe at socket welds. In most cases, the actual failure mechanism can not be conclusively determined due to weld joint damage. Therefore, cracking of small-bore stainless steel piping due to other causes will be treated as a detrimental aging effect. Cracking of small-bore stainless steel piping due to other causes is managed by the aging management program described in Section 4.3.3.

Operating event review procedures (Section 6.3) will identify small-bore cracking at other plants and permit plant specific actions to be taken to prevent similar events at Hatch, when it is determined that Hatch may be susceptible to the same concern.

Plant maintenance procedures (Section 6.2) provide for monitoring and trending vibration on major pumps and motors. Identifying excessive vibration and correcting it can prevent vibrational fatigue. When vibrational concerns are identified, the cause is determined and the design or installation is corrected. When a small-bore socket welded joint requires corrective action, whether this is due to a failure or because it has been identified as being susceptible to vibrational fatigue, it is repaired using a weld with a different configuration. There have been no instances of weld failure recurring following the specified weld repair at Plant Hatch.

Under the ISI program (Section 6.1) the small-bore piping is subject to a VT-2 inspection during the RRS pressure boundary testing per the requirements of Category B-P of ASME Section XI. Although volumetric inspection of small-bore piping welds is not effective and surface examination is not required for piping under 2 ½ inches, leakage detection requirements, coupled with the operating experience review procedures, maintenance procedures, and the plant QA program requirements for corrective actions, are adequate for 2 inch and smaller piping because of the low risk. Very small leaks can be detected, thus providing ample time to shutdown and repair an unisolable leak before the pipe breaks. The leak rate from RRS small-bore piping cannot exceed the makeup capacity and other plant systems ensure containment of any leakage. Because the maximum possible leak rate is low and multiple makeup sources exist, probabilistic risk assessment (PRA) analysis has shown this type event has a low contribution to core damage frequency.

5.2 Loss of Material

Aging mechanisms that can lead to loss of the material for the RRS Components are:

- Crevice corrosion
- Pitting corrosion

Crevice Corrosion

Crevice corrosion occurs in crevices or shielded areas that allow a corrosive environment to develop within the crevice. It requires stagnant conditions in the crevice and an oxygen content in the fluid above 100 ppb to initiate and perpetuate crevice corrosion. Although the oxygen content in crevices can differ significantly from the bulk fluid oxygen levels due to oxygen depletion, etc., a bulk fluid oxygen level to sustain the chemical reaction is necessary for the continued corrosion in the crevice.

Crevice corrosion is not expected to cause excessive degradation in crevice joints such as socket welds or flange joints in a properly controlled low impurity environment.

Stainless steel components in a treated water environment are not generally susceptible to crevice corrosion unless they are subject to stagnant or low flow conditions. The small-bore piping and valves are the only RRS components that are subject to low flow conditions. Therefore, loss of material due to crevice corrosion is a detrimental aging effect for the small-bore piping and valves, but not for any other RRS components. This aging effect is managed at Plant Hatch by the aging management program described in Section 4.3.4.

Pitting Corrosion

Pitting corrosion is more common with passive materials such as wrought austenitic stainless steels and nickel-base alloy steels than with non-passive materials. All nuclear plant materials are susceptible to pitting corrosion under certain conditions. Alloys containing molybdenum (e.g., Type 316 or 316L, CF-3M, and CF-8M) are somewhat more resistant to pitting. Oxygen levels above 100 ppb in conjunction with impurities such as chloride, fluoride, sulfate or copper are required to initiate pitting in carbon steel, low-alloy steel, wrought austenitic stainless steel, CASS, nickel-base alloys, and cast iron. Stagnant or low flow conditions which enables impurities to adhere to the metal surface is also required for pitting corrosion to occur. Areas where sludge piles and/or crevices exist are particularly susceptible to pitting corrosion.

Where crevice corrosion occurs in crevices that may contain stagnant fluid even under system flowing conditions, pitting requires either low flow or stagnant conditions to sustain the corrosion reaction and to provide for the concentration of contaminants. Maintaining an adequate flow rate will minimize pitting corrosion by preventing impurities from adhering to the material surface. The low flow threshold for stainless steel and treated water is < 3 ft/sec. The small-bore piping and valves are the only RRS components that are subject to low flow conditions. Therefore, loss of material due to crevice corrosion is a detrimental aging effect for the small-bore piping and valves, but not for any other RRS components. This aging effect is managed at Plant Hatch by the aging management program described in Section 4.3.4.

5.3 Loss of Fracture Toughness

Aging mechanisms that can lead to loss of fracture toughness are:

- Thermal Aging Embrittlement
- Neutron Irradiation Embrittlement (Not discussed in this report)

Thermal Aging Embrittlement

Thermal aging embrittlement degrades the mechanical properties of material (strength, ductility, and toughness) as a result of prolonged exposure to high temperatures. Carbon, low-alloy, cast iron, wrought austenitic stainless steel, copper alloys and nickel-base alloys are not susceptible to thermal aging embrittlement when exposed to normal nuclear plant operating environments. However, CASS materials are susceptible to thermal embrittlement. The degree of susceptibility is dependent upon material composition and exposure time at high temperatures. Castings with high ferrite and high molybdenum contents are more susceptible to thermal aging embrittlement than those with lower values.

Cast materials that are below the temperature-screening threshold of 482°F are not subject to significant reduction of fracture toughness for the period of extended operation. EPRI Report TR-106092 (Ref. 8) provides a methodology for determining if CASS components require aging management. Because the RRS pump casings (Section 4.2.3) and large-bore valve bodies (Section 4.2.4) are made from high-molybdenum, statically-cast material, loss of fracture toughness due to thermal aging embrittlement of these components is a detrimental aging effect that requires management at Plant Hatch by the aging management program described in Section 4.3.5.

5.4 Loss of Preload

Stress Relaxation

The pressure boundary bolting (Section 4.2.6) is susceptible to loss of preload due to stress relaxation. This aging effect only occurs under the conditions of constant strain where part of the elastic strain is replaced with plastic strain. The prestress in bolts or studs can relax with time at elevated temperatures. EPRI torquing procedures (Ref. 15) provide guidance in applying preload. Plant Hatch design requirements ensure that adequate preload is used for pressure boundary bolting. Maintenance procedures (Section 6.2) ensure this design preload is reapplied following maintenance. In the event that loose bolts are found, the maintenance procedures provide guidance from the EPRI torquing procedures for increasing the torque as needed. This aging effect is not considered a significant problem at Plant Hatch. However, a review of Plant Hatch pressure test results indicated several leaks in mechanical closures that may be due to loss of preload. Loss of preload can be minimized by ensuring that components are installed using applicable industry guidelines. However, simply applying good practices in the installation phase cannot prevent loss of preload. Therefore, loss of preload will be considered a detrimental aging effect at Plant Hatch. The aging management program for this aging effect is described in Section. 4.3.6.

6.0 AGING MANAGEMENT PROGRAMS

For this report, Aging Management Programs are divided into three categories. Monitoring Programs, Mitigating Programs, and Supporting Programs. This section will describe the programs in each of these categories.

6.1 Monitoring Programs

- Fatigue Monitoring Program
- Inservice Inspection Program (ISI)
- Technical Specification Surveillance Test Procedures

Fatigue Monitoring Program

This will be a new program at Plant Hatch. Equations are being developed to monitor the fatigue cumulative usage factor (CUF) for the locations that bound fatigue usage for the Class 1 piping 2 inches and larger. These equations are similar to the equations that have been used since 1985 at Plant Hatch to monitor fatigue CUF for the reactor pressure vessel. The existing program for the Hatch RPV is described in Section 4.2 of the Unit 1 FSAR and Section 5.4 of the Unit 2 FSAR.

The new program for Class 1 piping will periodically require that specified plant transients be totaled since the last time the fatigue usage was assessed. This will be done at least once a year. The formulas will then be used to calculate the increase in fatigue usage and establish the new fatigue CUF for each of these locations. The acceptance criteria will be a CUF less than 1.0. Based on the number of transients over the last 5 to 10 years, projections can be made for the fatigue CUF after 60 years of operation. For example, if 6 years is chosen, the applicable transients over the last 6 years would be totaled and then averaged. The average number of transients would then be multiplied by the remaining years of the 60-year life. This projected number of future transients would be multiplied by the fatigue monitoring formula coefficient for that transient to determine the projected increase in fatigue CUF and that value added to the current value of fatigue CUF. The reactor pressure vessel sees similar cycles to those seen by the RRS piping. The 40-year CUF based on design transients is much less than 1.0 and industry experience is the CUF determined based on actual cycles is usually less than the CUF based on design transients. Therefore, it is expected that this projection will show that the fatigue CUF for all locations will remain less than 1.0 through the extended operating period. This projection will be performed when the program is established.

The scope of this program will include all of the Class 1 piping except piping 1 inch and smaller that was designed to Class 2 and 3 criteria for a minimum of 7,000 cycles.

The design fatigue CUF of the RRS piping is low. The 40-year design calculation for the bounding location is 0.0986, which is a TLAA dispositioned in Section 7.2 of this report. Assuming linear extrapolation of this calculation to 60 years, the value would only be 0.1479. The BWR Primary Coolant Pressure Boundary License Renewal Industry Report (Ref. 5) states, "For fatigue to be resolved, the component must have no experience of flaws induced by fatigue and either typical evaluations of the component with similar design and operation or plant-specific evaluations show that the fatigue usage factor is less than 0.4 for SS or 0.25 for CS for at least 40 years." Therefore, it would be reasonable to exclude any location with a projected fatigue CUF less than 0.40 for 60 years from the Fatigue Monitoring Program. However, Plant Hatch will not do this unless the 60-year projection based on actual plant operating history is also less than 0.40 once the Fatigue Monitoring Program is established.

If the acceptance criteria are exceeded, corrective actions will be reanalysis, flaw (crack) growth analysis, more frequent ISI inspections, or replacement. Administrative controls provide for formal review and approval of changes to the controlling procedures. This program will be an Appendix B program subject to evaluation by the NRC, industry, and site organizations. More details will be available on this program once the formulas are developed and the program is established.

Inservice Inspection Program (ISI)

All plant Class 1, 2, and 3 components, including the RRS stainless steel piping, valve bodies and pressure boundary bolting are in the scope of the plant's ISI program. The program includes both ASME Code requirements as well as augmented inspections required by regulation. The purpose of the program is to inspect plant components in a manner and to a schedule that is adequate, considering the components safety significance, to ensure degradation of the component is found and corrected before there is a loss of safety function.

The large-bore Class 1 piping (Section 4.2.1) welds, including the welds between the pipe and the pump casing and between the pipe and the valve bodies, periodically receive volumetric and surface examinations per ASME Section XI, Table 2500-1, Category B-J for Class 1 piping. These inspections include the most likely locations for cracking: the stainless steel piping weldments and the heat affected zone on either side of the weld. The regulatory direction contained in Generic Letter 88-01 (Ref. 9) and NUREG-0313 (Ref. 4) is also applied to this piping along with the additional guidance provided via General Electric Nuclear Energy in SIL No. 117 (Ref. 10). Although the purpose of this additional guidance is to inspect for cracking due to IGSCC (Section 5.1), cracking due to fatigue (Section 5.1) or loss of fracture toughness because of thermal aging embrittlement (Section 5.2) would also be found, should it exist.

Pipe geometry prevents volumetric exams from being effective for smaller pipe. Welds in piping 2 ½ to 4 inches in diameter only receive surface examinations. Pipe 2 inches and smaller is exempt from this requirement. This is acceptable for three reasons. The time from crack initiation to through wall propagation of small-bore piping (Section 4.2.2) is less than one operating cycle so this inspection is not effective. Leak detection allows the existence of very small leaks to be discovered so there is ample opportunity to shutdown and repair them before there is a major break. The system makeup capacity exceeds the maximum leak rate from 2 inch and smaller piping.

All the RRS commodities are subject to a VT-2 inspection during the RRS pressure boundary testing per the requirements of Category B-P of ASME Section XI. This inspection is performed near the end of an outage to ensure that unacceptable leakage is identified and corrected before returning to operation.

If they are disassembled for maintenance, the RRS pump casings (Section 4.2.3) and valve bodies (Section 4.2.4) are subject to a VT-3 inspection per the requirements of ASME Section XI, Categories B-L-2 and B-M-2 for Class 1 components. When conditions are detected during the pump casing or valve body inspections that exceed the allowable limits, engineering evaluations (such as crack growth analyses) of detected flaws are carried out to determine acceptability of the flaws, or to establish the time for corrective actions.

Class 1 Bolting is subject to inspection every ten years. The studs from one pump casing receive a volumetric examination once every ten years per the requirements of ASME Section XI, Category B-G-1. The nuts from one pump and a sample of other Class 1 bolts and nuts receive a VT-1 visual inspection once every ten years per the requirements of ASME Section XI, Categories B-G-1 and B-G-2.

The methods of detection are prescribed in the ASME code and the applicable plant procedures. The frequency of the inspections for the RRS commodities complies with the ASME Section XI Code requirements and considers the installed materials, aging effects, reactor coolant water properties (including the use of HWC (Section 6.2) and electrochemical corrosion potential (ECP) measurements). The ISI program provides sample sizes, acceptance criteria, and corrective actions, and requires confirmation that those actions have been effective. The program also provides for monitoring and trending for timely corrective actions.

The plant is currently committed to the 1989 Edition of the ASME Section XI Code. The plant is currently required by regulation to upgrade to a newer version of the code approximately every 10 years. An addendum is issued annually with incorporation of any addenda into a base code every 3 years. ISI Inspections consist of visual, surface, and volumetric inspections as well as leakage inspection in conjunction with pressure testing. The plant ISI program is required by regulation. It well known and understood by the NRC and has been evaluated by the NRC, industry, and site organizations. The acceptability of these inservice examination program elements is supported by the evaluation findings in NUREG-0800 (Ref. 20), which states, in part,

> The conduct of periodic inspections and leakage and hydrostatic testing of pressure-retaining components of the reactor coolant pressure boundary in accordance with the requirements of Section XI of the ASME Code provides reasonable assurance that evidence of structural degradation or loss of leak-tight integrity occurring during service will be detected in time to permit corrective action before the safety function of a component is compromised.

Technical Specification Surveillance Procedures

Plant Hatch Technical Specifications include requirements for testing and inspecting safety related components to ensure their safety function is maintained. These tests and inspections are performed using surveillance procedures. Some of these tests and inspections verify requirements in the plant's Technical Specifications are met. Others verify specifications in the plant Hatch Technical Requirements Manual (TRM) are met. Requirements in the TRM are similar to Technical Specification requirements, but they have a lower level of control in that NRC approval is not required to change them. The scope of these surveillance procedures includes requirements for monitoring Reactor Coolant System (RRS) leak rate and chemistry.

Technical Specification surveillance procedures are used to monitor the type and quantity of RRS system leakage. The Bases for the Plant Hatch Technical Specification B3.4.4 states,

> The allowable RCS operational LEAKAGE limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background LEAKAGE due to equipment design and the detection capability of the instrumentation for determining system LEAKAGE were also considered. The evidence from experiments suggests that, for LEAKAGE even greater than the specified unidentified LEAKAGE limits, the probability is small that the imperfection or crack associated with such leakage would grow rapidly.
>
> The unidentified LEAKAGE flow limit allows time for corrective action before the RCPB could be significantly compromised. The 5 gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs (Refs. 4 and 5) shows that leakage rates of hundreds of gallons per minute will precede crack instability (Ref. 6).

Violation of leakage limits in Technical Specification 3.4.4 is considered to be a Limiting Condition for Operation (LCO). If excessive leakage from unidentified sources were to occur during operation, the surveillance procedures require the source of leakage be identified. If the leakage cannot be identified within a time limit or if any leakage is identified as pressure boundary leakage, the unit must be shutdown.

TRM water chemistry requirements mitigate loss of material due to crevice corrosion and pitting corrosion (Section 5.2) and cracking due to IGSCC (Section 5.1) and other mechanisms. Table T.3.4.1.1 in the TRM limits chloride concentration and conductivity of the RRS water. If the table limits are exceeded and not restored within the allowed time, the unit must be shutdown.

These procedures provide methods of detection, frequency of monitoring, acceptance criteria, and corrective actions. If corrective actions are required because a Technical Specification surveillance test is unsatisfactory, the surveillance test must be reperformed satisfactorily following completion of the corrective actions. Administrative controls are in place to provide a formal review and approval process for changes to surveillance procedures. The plant surveillance program is an Appendix B program required by regulation. It is well known and understood by the NRC and has been evaluated by the NRC, industry, and site organizations.

6.2 Mitigating Programs

- Hydrogen Water Chemistry (HWC)
- Induction Heat Stress Improvement (IHSI)
- Maintenance Procedures
- Mechanical Stress Improvement Process (MSIP)
- Water Chemistry Procedures
- Weld Overlays

Hydrogen Water Chemistry (HWC)

ECP is a measure of the oxidizing power of a solution and is a driving force for IGSCC (Section 5.1). The higher the ECP, the greater the probability of initiation and growth of cracking in austenitic stainless steels. Short-term tests have shown that IGSCC is not observed when the ECP is below -230 mV(SHE). IGSCC initiation is observed in sensitized steel when the ECP is above -230 mV (SHE), even in reactor water with low conductivity (a measure of impurities).

The addition of excess hydrogen to the feedwater (HWC) reduces the production of oxygen and peroxide by radiolysis in the core region. At Hatch, protection of the recirculation piping is achieved with lower injection rates than is needed for protection of the reactor internals. Moderate HWC (45 - 50 SCFM, or about 1.5 ppm hydrogen in the feedwater) is required for protection of the bottom head drain and core plate, which is the reactor internals area targeted for protection at Plant Hatch.

Plant Hatch has begun noble metal addition of Platinum and Rhodium on Unit 1during the Spring, 1999 outage. This will result in the same or greater protection with less hydrogen addition, which will also reduce dose rates during plant operation. Noble metal addition is planned to be used on Unit 2 starting with the next outage and is expected to be an ongoing program for both units.

At Plant Hatch, HWC controls the ECP values and thus reduces the oxygen and peroxide content and thereby, mitigates the corrosive environment in the RRS. The purpose of HWC is to prevent crack initiation and growth due to IGSCC. Plant and industry operating experience were significant contributors to development of this process. HWC at Plant Hatch has been evaluated by the NRC, industry, and site organizations.

Induction Heat Stress Improvement (IHSI)

IHSI converts the weld joint residual stresses from tensile to compressive stress, which reduces the susceptibility of the weld joint to IGSCC (Section 5.1). IHSI was used for RRS large bore piping welds on both units. The process is performed using approved plant procedures that are reviewed and approved by a formal process. This process has been evaluated by the NRC, industry, and site organizations.

Maintenance Procedures

Plant Hatch maintenance procedures provide direction and guidance for plant maintenance activities including preventive maintenance, inspections, and repairs. Welding procedures, bolt torquing procedures, procedures for disassembly, inspection, and repair of RRS components are included.

The weld procedures include a weld repair for 1 inch and smaller socket welds that have failed, or have been identified as at risk to fail. There have been no instances of weld failure recurring following the specified weld repair at Plant Hatch.

These procedures are reviewed and approved by a formal process. Plant maintenance activities are routinely evaluated by the NRC, industry, and site organizations.

Mechanical Stress Improvement (MSIP)

MSIP uses a hydraulic system to uniformly compress the entire pipe at a location near the weld joint. MSIP converts the weld joint residual stresses from tensile to compressive stress, which reduces the susceptibility of the weld joint to IGSCC (Section 5.1). MSIP was used for some of the RRS large bore piping welds on both units. The process is performed using approved plant procedures that are reviewed and approved by a formal process. This process has been evaluated by the NRC, industry, and site organizations.

Water Chemistry Procedures

Plant Hatch water chemistry procedures establish limits for the water chemistry for water from various sources as well as methods and frequency of sampling. These procedures implement the recommendations of the EPRI Water Chemistry Guidelines (Ref. 19). The scope of these procedures includes the water in RRS stainless steel piping, pumps, and valves. The purpose of these procedures is to control the chemistry of fluid systems at the plant and thereby mitigate degradation of the system components. Parameters are monitored at least daily in the RRS and include conductivity, chloride, pH, silica, sulfate, and ECP. Cracking due to IGSCC and loss of material due to crevice corrosion or pitting corrosion are the primary aging effects mitigated in the RRS, although other mechanism and effects are also mitigated by controlling these parameters. These aging effects, if left unchecked, could degrade RRS components to the point that the primary pressure boundary function would no longer be maintained. The procedures specify acceptance criteria, and in most cases, there are multiple action levels specified. For each action level, there is a required corrective action. Corrective actions include root cause determination to ensure prevention of recurrence, when appropriate. Water chemistry procedures are reviewed and approved by a formal process. As industry or plant experience shows it to be advisable, the limits are modified. During past six years there have been a total of only five reactor water chemistry events, at Plant Hatch, where the sulfate, chloride and/or conductivity limit for Action Level 1 has been exceeded. The technical evaluations conducted following these events concluded that no significant crack initiation or crack growth in plant materials of construction will result. The water chemistry procedures are governed by Appendix B requirements and have been evaluated by the NRC, industry, and site organizations.

Weld Overlays

Because of cracking due to IGSCC (Section 5.1) in the RRS piping, the Unit 1 large-bore piping (Section 4.2.1) in this system was repaired using weld overlays. The weld overlays are made of ER308L grade stainless steel, an IGSCC resistant material. The filler metal for weld overlay repair of Inconel to stainless steel welds are Inconel 82, also an IGSCC resistant material. All weld overlay designs were prepared assuming a bounding 360 degree circumferential oriented crack through wall flaw in accordance with the requirements of the "Standard Weld Overlay" design intended for indefinite service per NUREG 0313, Revision 2 (Ref. 4). To date, none of the existing cracks have propagated into the overlay. The overlays are governed by Appendix B requirements and have been evaluated by the NRC, industry, and site organizations.

6.3 Supporting Programs

- Quality Assurance (QA) Program
- Operating Event Review Procedures

Quality Assurance (QA) Program

The QA program provides control over activities affecting the quality of structures, systems, and components consistent with their importance to safety. Activities that are controlled by the QA Program include but are not limited to:

- Administrative controls ensure that important activities are performed in accordance with procedures.
- Procedure control and review provides a formal process for review and approval of plant procedures and changes to those procedures.
- Corrective action program ensures that timely corrective actions are taken including requirements for root cause investigations and actions to prevent recurrence.
- Inservice inspection program (Section 6.1).
- Surveillance program (Section 6.1).
- Maintenance programs (Section 6.2).
- Plant chemistry (Section 6.1).

The Plant Hatch QA program applies to all plant activities and commodities discussed in this report and assures they meet appropriate sections of 10 CFR 50, Appendix B, criteria. The QA program has been evaluated by the NRC, industry and site organizations.

Operating Event Review Procedures

Plant Hatch has procedures in place to review plant and industry operating experience events. These reviews provide for corrective actions to prevent recurrence of significant plant events. Corrective actions are also taken to prevent industry events that are determined to be applicable to Plant Hatch. Sources of industry events include EPIX, GE Service Information Letters, INPO, EPRI, NEI, and NRC communications.

7.0 EVALUATION OF TIME LIMITED AGING ANALYSES

7.1 TLAA Issues and Exemptions That Require Evaluation

The Plant Hatch process (Ref. 2) was used to identify time-limited aging analysis (TLAA) issues for the RRS. A review of the design calculations and evaluations was conducted to identify potential issues. The six criteria delineated in paragraph 54.3 of the Rule (Ref. 1) were used and it was determined that the following TLAA issue applies to the RRS pressure boundary components. This issue is evaluated in Section 7.2.

- Evaluation of the large-bore stainless steel piping fatigue usage for the extended operating period,

A review of the relevant licensing correspondence identified that there are no exemptions granted under 10 CFR 50.12 that are in effect and based on TLAA issues.

7.2 Evaluation of the Time Limited Aging Analysis Issues

Fatigue

An explicit fatigue (Section 5.1) analysis was performed for the large-bore piping on both units as part of power uprate. Considering both units, for the most limiting location in the system piping and using the current licensing basis cyclic duty, the fatigue CUF at the end of a 40-year life was calculated to be 0.0986. Extending this usage factor linearly to a 60-year life yields a CUF of 0.1479 at the most limiting location. Therefore, the extrapolated CUF for the large-bore, stainless steel piping components are significantly less than 1.0. The ASME Code, Section XI, has established that cracking due to fatigue will not occur at CUF values below 1.0. Generic evaluations described in References 11, 12, and 13 demonstrate that there is adequate margin in the existing ASME Code Section III explicit fatigue design process to account for reactor water environmental effects.

The original Plant Hatch CLB required that a break point be postulated at any piping location where the projected cumulative usage factor exceeds 0.1. For the extended operating period, some locations with a 40-year CUF less than 0.1 will have a 60-year CUF greater than 0.1. The locations where this condition exists have been identified along with several possible methods of resolution. The method of resolution that will be used will be specified in the renewal application.

Plant Hatch will not revise the Class 1 piping fatigue calculations. The fatigue TLAA for large-bore stainless steel piping (Section 4.2.1) in the RRS, is dispositioned by establishing an aging management program as described in Section 4.3.1 to monitor fatigue of the large-bore stainless steel piping using the Fatigue Monitoring Program described in Section 6.1.

Class 1 small-bore piping (Section 4.2.2) 1 inch and under in diameter is designed for 7,000 full-temperature cycles. For the RRS, the system piping will not exceed 7,000 cycles during 60 years of operation. Therefore, the calculations for this piping remain valid for the extended operating period. No revision to the calculation is required and no aging management program is needed to disposition the fatigue TLAA for small-bore RRS piping.

8.0 TECHNICAL SPECIFICATION CHANGES OR ADDITIONS

The Plant Hatch technical specifications were reviewed to determine if they are affected by the results of the aging management review in Section 4.0. The review determined that the technical specifications would not need to be changed.

9.0 REFERENCES

Entire publications referenced in this report are not to be incorporated into the Plant Hatch current licensing basis (CLB). Each publication referenced is only to be included in the CLB to the extent it relates to the specific area discussed in the portion of this report that the document is referenced.

1. 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," 60 FR 22491, May 8, 1995.

2. "Edwin I. Hatch Nuclear Plant License Renewal Process Methodology Document," April 13, 1998.

3. NEI 95-10, "Revision 0, Industry Guideline on Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," March 1996.

4. NUREG-0313, "Technical Report on Material Selection and Processing guidelines for BWR Coolant Pressure Boundary Piping," Revision 2, January 1998.

5. EPRI Report TR-103843, "BWR Primary Coolant Pressure Boundary License Renewal Industry Report," Revision 1, July 1994.

6. Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, Working Draft, September, 1997.

7. ASTM STP 756 – "Intergranular Stress-Corrosion Cracking Resistance of Austenitic Stainless Steel Castings," 1982

8. EPRI Report TR-106092, "Evaluation of Thermal Aging Embrittlement for Cast Austenitic Stainless Steel Components," April 1997.

9. Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," January 1988 and Supplement 1, February 1992.

10. General Electric Nuclear Energy Service Information Letter (SIL) No. 117

11. EPRI Report TR-107943, "Environmental Fatigue Evaluations of Representative BWR Components," June 1998.

12. EPRI Report TR-110356, "Evaluation of Environmental Thermal Fatigue Effects on Selected Components in a BWR Plant," April 1998.

13. EPRI Report TR-107521, "Generic License Renewal Technical Issues Summary," Final Report, April 1998.

14. General Electric Service Information Letter (SIL) No. , Supplement 1, "Byron-Jackson Recirculation Pump Heat Exchanger Leakage," March 23, 1990

15. EPRI Bolting Procedures Reference Manual, NP5067, Vol. 1, "A Reference Manual for Nuclear Power Plant Maintenance Personnel, Large Bolt Manual."

16. Pressure Vessel Research Council WRC Bulletin 1993, Nuclear Piping Criteria for Advance Light Water Reactors, "Failure Mechanisms and Corrective Actions."

17. NRC AEOD/E308 Report, 4/19/83 "Cracks and Leaks in Small Diameter Piping."

18. BAW-2243A, "Demonstration of the Management of Aging Effects for the Reactor Coolant System Piping", The B&W Owners Group Generic License Renewal Program, June 1996.

19. EPRI Report TR- 103515-R1, "BWR Water Chemistry Guidelines - 1996 Revision," BWRVIP-29, 2493, Final Report, December 1996.

20. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" June 1987.

10.0 DEFINITIONS

1. **BWR** – Boiling Water Reactor

2. **CASS** – Cast Austenitic Stainless Steel

3. **CLB** – Current Licensing Basis

4. **CUF** – Cumulative Usage Factor

5. **Detrimental Aging Effect** – Plausible aging effects that are determined to be applicable for Plant Hatch and could cause sufficient degradation of the SCC to potentially result in a loss of the intended function

6. **ECP** – Electrochemical Corrosion Potential

7. **FSAR** – Final Safety Analysis Report

8. **HWC** – Hydrogen Water Chemistry

9. **IHSI** – Induction Heating Stress Improvement

10. **IPA** – Integrated Plant Assessment

11. **LCO** – Limiting Condition for Operation

12. **LPCI** – Low Pressure Coolant Injection

13. **PCPB** – Primary Coolant Pressure Boundary

14. **Plausible Aging Effect** – Aging effect with evidence of possibly occurring in the industry for the material and environment under consideration for Plant Hatch.

15. **PRA** – Probability Risk Assessment

16. **RHR** – Residual Heat Removal

17. **RPV** – Reactor Pressure Vessel

18. **RRS** – Reactor Recirculation System

19. **RWCU** – Reactor Water Cleanup

20. **SCC** – Structure, Component, or Commodity

21. **SSC** – System, Structure, or Component

22. **TLAA** – Time Limited Aging Analysis

February 3, 2000

MEMORANDUM TO: Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs, NRR

FROM: Raj K. Anand, Project Manager ***/RA/***
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs, NRR

SUBJECT: MEETING WITH SOUTHERN NUCLEAR OPERATING COMPANY (SNC) ON LICENSE RENEWAL FOR HATCH, UNITS 1 AND 2

DATE & TIME: Thursday, February 10, 2000
8:30 a.m. - Overview of SNC's license renewal application relative to 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

1:00 - 3:00 p.m. - Overview of SNC's environmental report relative to 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions"

LOCATION: U.S. Nuclear Regulatory Commission
Two White Flint North
11545 Rockville Pike
Rockville, Maryland
Room T-2 B3

PURPOSE: To discuss the SNC's license renewal application for Hatch Units 1 and 2.

PARTICIPANTS:*

NRC	SNC
R. Zimmerman, NRR	S. Long
J. Johnson, NRR	R. Baker
J. Strosnider, NRR	et al.
S. Newberry, NRR	
C. Carpenter, NRR	
C. Grimes, NRR	
R. Wessman, NRR	
W. Burton, NRR	
et al.	

Docket Nos. 50-321 and 50-366

cc w/encl: See next page

CONTACT: Raj K. Anand, NRR

12/17

301-415-1146

*Meetings between NRC technical staff and applicants or licensees are open for interested members of the public, intervenors, or other parties to attend as observers pursuant to " Commission Policy Statement on Staff Meetings Open to the Public" 59 Federal Register 48340, 9/20/94.

February 3, 2000

MEMORANDUM TO: Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs, NRR

FROM: Raj K. Anand, Project Manager /RA/
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs, NRR

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Docket Nos. 50-321 and 50-366
cc w/encl: See next page
CONTACT: Raj K. Anand, NRR
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OPA
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Edwin I. Hatch Nuclear Plant

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 22, 2000

LICENSEE: SOUTHERN NUCLEAR OPERATING CO., INC.

FACILITY: EDWIN I. HATCH NUCLEAR PLANT, UNIT Nos. 1 AND 2

SUBJECT: SUMMARY OF MEETING WITH SOUTHERN NUCLEAR OPERATING CO., INC., REGARDING HATCH UNITs 1 AND 2 LICENSE RENEWAL APPLICATION

On February 10, 2000, representatives of Southern Nuclear Operating Co., Inc. (SNC), met with the Nuclear Regulatory Commission (NRC) staff to provide an overview of its license renewal application to be submitted in accordance with 10 CFR Part 54 requirements and the associated environmental report submitted in accordance with 10 CFR Part 51 requirements. The meeting was divided into two sessions with discussions regarding the renewal application in the morning and discussions on the environmental report in the afternoon. Attendees at the two sessions are listed in Attachments 1 and 2. Presentation materials used by SNC are contained in Attachments 3 and 4.

SNC indicated that it plans to submit the renewal application for the Edwin I. Hatch Nuclear Plant, Unit Nos. 1 and 2 (Hatch) in February 2000. The application will use a "commodity" approach to group and review in-scope structures and components. SNC is monitoring the ongoing review of the Calvert Cliffs and Oconee renewal applications and the resolution of generic renewal issues and has incorporated lessons learned into the Hatch application. Two drafts of the application were peer reviewed and peer feedback was incorporated into the application.

The reactor vendor for Hatch is General Electric. SNC will utilize the generic and relevant information from the Boiling Water Reactor Vessel Internals Project license renewal topical reports.

SNC provided an overview of its organization, its approach to license renewal, and the results of its environmental review. Using the presentation materials, SNC summarized the format and content of its application. SNC's presentation was beneficial as it provided the staff with an overview of the Hatch application which will facilitate the staff's review when the application is received.

William Burton

William F. Burton, Senior Project Manager
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Attachments: As stated

cc w/atts: See next page

B/8

Souther Nuclear Operating Company
Edwin I. Hatch Nuclear Plant Units 1 and 2

Docket Nos.: 50-321, 50-366

cc:

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ATTENDANCE LIST
EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
GENERAL AND TECHNICAL OVERVIEW
FEBRUARY 10, 2000

	NAME	ORGANIZATION
1.	William Burton	NRC/NRR/DRIP/RLSB
2.	Hai-Boh Wang	NRC/NRR/DRIP/RLSB
3.	Winston Liu	NRC/NRR/DRIP/RLSB
4.	Raj Anand	NRC/NRR/DRIP/RLSB
5.	Steve Hoffman	NRC/NRR/DRIP/RLSB
6.	Jim Strnisha	NRC/NRR/DRIP/RLSB
7.	P.T. Kuo	NRC/NRR/DRIP/RLSB
8.	Sam Lee	NRC/NRR/DRIP/RLSB
9.	Chris Grimes	NRC/NRR/DRIP/RLSB
10.	Bhagwat Jain	NRC/NRR/DE/EMEB
11.	John Fair	NRC/NRR/DE/EMEB
12.	Francis Grubelich	NRC/NRR/DE/EMEB
13.	John Ma	NRC/NRR/DE/EMEB
14.	David Jeng	NRC/NRR/DE/EMEB
15.	Hans Ashar	NRC/NRR/DE/EMEB
16.	Y.C. (Renee) Li	NRC/NRR/DE/EMEB
17.	Arnold Lee	NRC/NRR/DE/EMEB
18.	Pei-Ying Chen	NRC/NRR/DE/EMEB
19.	Kamal Manoly	NRC/NRR/DE/EMEB
20.	Y.S. Kim	NRC/NRR/DE/EMEB
21.	Duc Nguyen	NRC/NRR/DE/EEIB
22.	Amar Pal	NRC/NRR/DE/EEIB
23.	Paul Shemanski	NRC/NRR/DE/EEIB
24.	George Georgiev	NRC/NRR/DE/EMCB
25.	C.E. Carpenter	NRC/NRR/DE/EMCB
26.	Timir Misra	NRC/NRR/DE/EMCB
27.	Matthew Mitchell	NRC/NRR/DE/EMCB
28.	Carolyn Lauron	NRC/NRR/DE/EMCB
29.	James Davis	NRC/NRR/DE/EMCB
30.	Pat Patnaik	NRC/NRR/DE/EMCB
31.	William Koo	NRC/NRR/DE/EMCB
32.	Kris Parczewski	NRC/NRR/DE/EMCB
33.	Louise Lund	NRC/NRR/DE/EMCB
34.	Lee Banic	NRC/NRR/DE/EMCB
35.	Bill Bateman	NRC/NRR/DE/EMCB
36.	Keith Wichman	NRC/NRR/DE/EMCB
37.	Barry Elliot	NRC/NRR/DE/EMCB
38.	Allen Hiser	NRC/NRR/DE/EMCB
39.	Ted Sullivan	NRC/NRR/DE/EMCB

Attachment 1

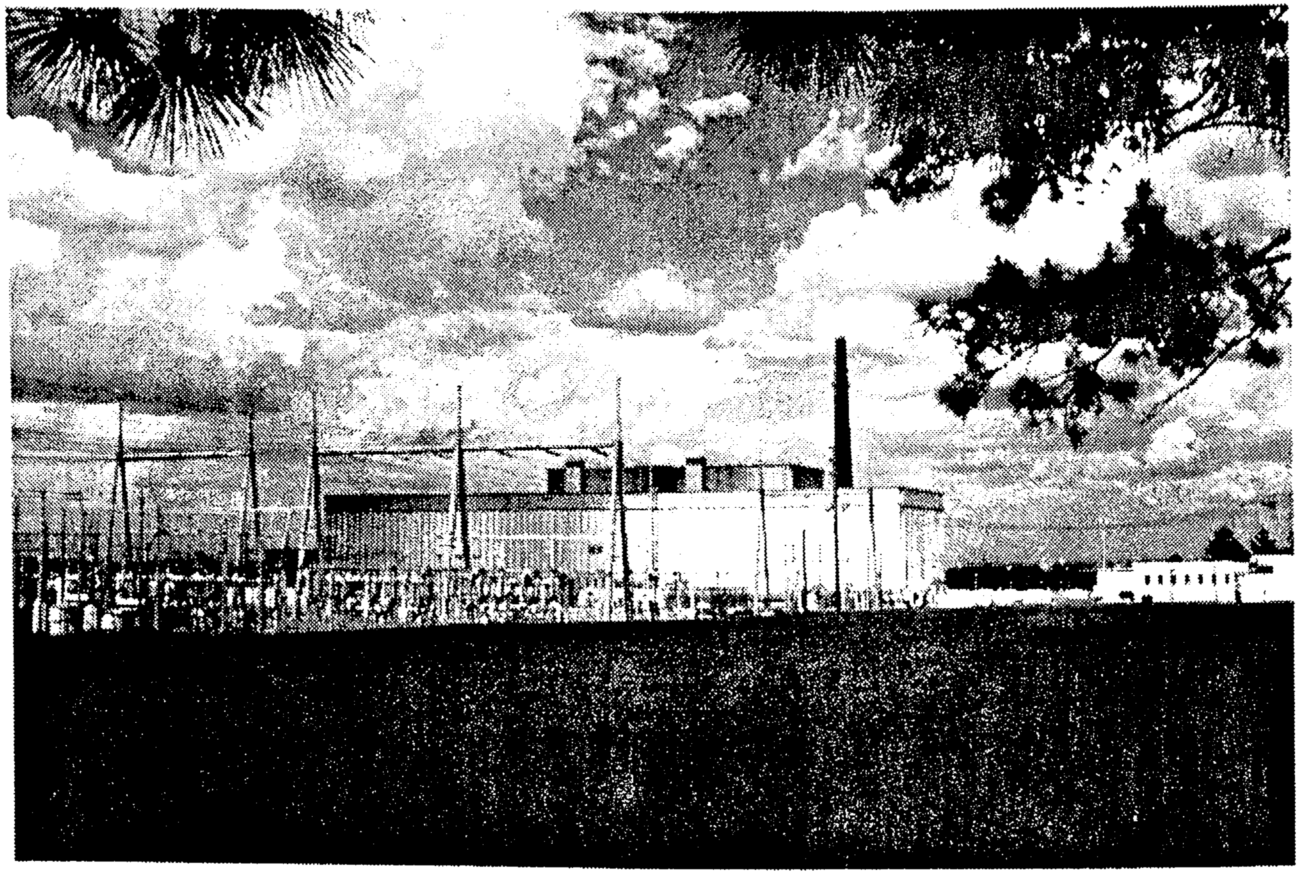
40.	Meena Khanna	NRC/NRR/DE/EMCB
41.	Stephanie Coffin	NRC/NRR/DE/EMCB
42.	Robert Hermann	NRC/NRR/DE/EMCB
43.	Scott Kirk	Southern Co.
44.	Ray Baker	Southern Co.
45.	Louis Long	Southern Nuclear
46.	James Davis	Southern Nuclear
47.	Charles Pierce	Southern Nuclear
48.	Don Palmrose	NUS-LIS
49.	Len Olshan	NRC/NRR/DLPM
50.	Kathryn Sutton	Winston Strawn
51.	Tatsuya Taminami	Tokyo Electric
52.	Goutam Bagchi	NRC/NRR/DE
53.	Mark Crisler	SCS
54.	Muhammad Razzaque	NRC/NRR/DSSA/SRXB
55.	Caudle Julian	NRR/R II

ATTENDANCE LIST
EDWIN I. HATCH NUCLEAR PLANT, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
ENVIRONMENTAL OVERVIEW
FEBRUARY 10, 2000

	NAME	ORGANIZATION
1.	James Luehman	NRC/NRR/DRIP/RGEB
2.	Barry Zalcman	NRC/NRR/DRIP/RGEB
3.	Thomas Kenyon	NRC/NRR/DRIP/RGEB
4.	Robert Jolly	NRC/NRR/DRIP/RGEB
5.	William Burton	NRC/NRR/DRIP/RLSB
6.	Jim Strnisha	NRC/NRR/DRIP/RLSB
7.	Steve Hoffman	NRC/NRR/DRIP/RLSB
8.	Chris Grimes	NRC/NRR/DRIP/RLSB
9.	Ray Baker	Southern Co.
10.	Scott Kirk	Southern Co.
11.	Mark Crisler	Southern Co.
12.	Charles Pierce	Southern Nuclear
13.	Louis Long	Southern Nuclear
14.	James Davis	Southern Nuclear
15.	Kathryn Sutton	Winston Strawn
16.	Mary Ann Parkhurst	PNNL
17.	Janice Moore	NRC/OGC
18.	Robert Palla	NRC/NRR/DSSA/SPSB

Attachment 2

Plant Hatch
License Renewal Application



SOUTHERN COMPANY

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Agenda

- *Introduction*
- *Background and Objective*
- *Plant Hatch License Renewal Project Overview*
- *License Renewal Application Contents*

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Introduction

Who we are -

- SNC is a wholly-owned subsidiary of Southern Company, the largest producer of electricity in the U.S.
- The licensee operator and agent for the owners of Plants Farley, Hatch, and Vogtle

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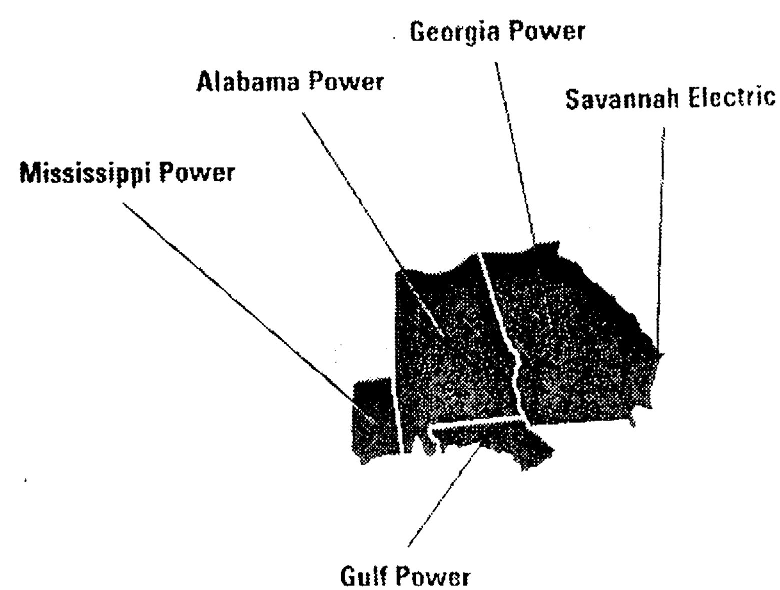
Introduction

Traditional Business Area



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Introduction

Our industry participation -

- SNC has maintained participation in BWR Owners Group license renewal activities since 1989 and WOG license renewal activities since 1992

- SNC participates in the NEI and EPRI License Renewal efforts
  - Ongoing activities include resolving generic license renewal issues, work on revising NEI 95-10 (industry guidance for license renewal), input to update of the Standard Review Plan, the GALL report, etc.

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Introduction

Utilize monthly management meetings to maintain SNC and NRC focus on process

– ***Continue meetings similar to the Calvert Cliffs/NRC meetings to***

- *brief NRC line management on progress of license renewal application review*
- *determine if any issues need elevating to NRC Steering Committee by SNC or NRC*
- *establish performance indicators for next review meeting, such as schedule adherance, quality of work, communications, regulatory stability, accomplishments, and concerns*

Background, Objective and Project Overview

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Background

- Unit 1's current license expires in August 2014
- Unit 2's current license expires in June 2018
- 10 CFR 54 allows the issuance of a renewed license for an additional 20 years
- SNC plans to request renewal of the licenses by filing an application in February 2000
- When approved, the renewed licenses would allow continued operation until August 2034 and June 2038 for Units 1 and 2 respectively

Objective

Obtain a renewed license for Plant Hatch in a timely and efficient manner based on:

- *Utilize the generic and relevant information from the BWRVIP Topical Reports*
- *Implementing relevant lessons learned from the Calvert Cliffs and Oconee application review process*
- *Improved guidance resulting from the Calvert Cliffs and Oconee experience*
- *Coordinating with NRC to help ensure the Hatch application preparation and review process are effective, efficient, timely, and predictable*

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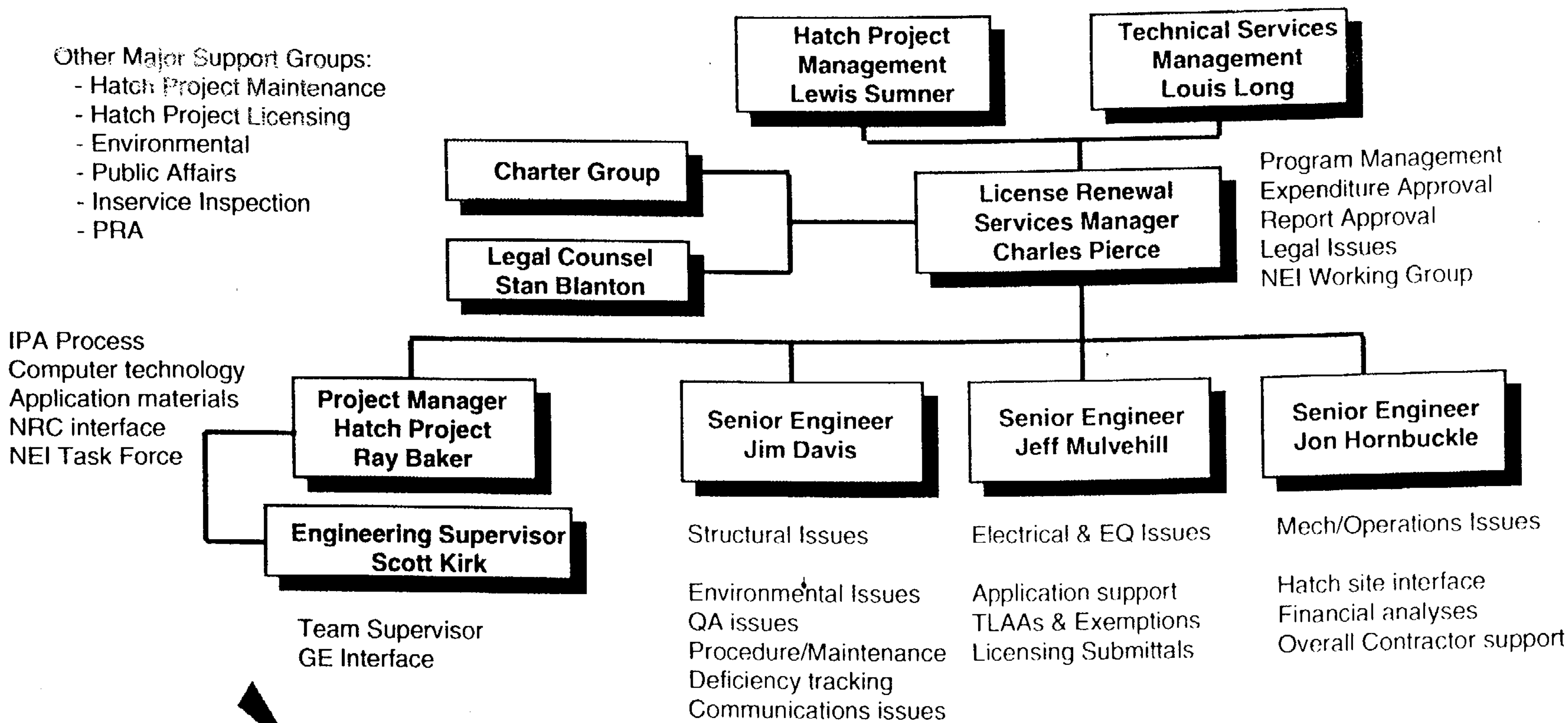
Project Overview

Our project organization -



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Project Overview

Application Approach -

- Participated in NEI/EPRI issue resolution and incorporated relevant results
- Monitored Calvert Cliffs and Oconee interactions with the NRC & incorporated relevant information
- Used NRC's new standard format for the preparation of the Hatch Application
- Conducted peer reviews of two drafts of the application and incorporated input

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Project Overview

Assembling the application - what we did -

- *Provided general information pursuant to §54.19*
  - *General information specified in 10 CFR 50.33(a) - (e), (h), and (i) {§54.19(a)}*
  - *Included conforming changes to the standard indemnity agreement to account for the expiration term of the proposed renewed license {§54.19(b)}*

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Project Overview

Assembling the application - what we did -

- **Performed an Integrated Plant Assessment pursuant to §54.21(a)**
  - *Identified and listed structures and components that are subject to an aging management review per §54.4(a) {§54.21(a)(1)}*
  - *Described and justified the scoping and screening methodology used to identify and list {§54.21(a)(2)}*
  - *Demonstrated for each structure and component subject to aging management review, that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation {§54.21(a)(3)}*

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Project Overview

Assembling the application - what we did -

- **Evaluated Time-Limited Aging Analyses (TLAAs) pursuant to §54.21(c)**
  - *Identified and listed calculations and analyses that meet the criteria for TLAAs and dispositioned the TLAAs per one of the three demonstrations of §54.21(c)(1)(i)-(iii)*
  - *Identified and listed plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on time-limited aging analyses {§54.21(c)(2)} --- there were none*

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Project Overview

Assembling the application - what we did -

- Produced an FSAR supplement pursuant to §54.21(d)
  - Provided summary descriptions of programs and activities for managing the effects of aging
  - Provided a summary of the evaluations of TLAAs for the period of extended operation
- Provided Technical Specifications changes for the renewal term pursuant to §54.22
- Produced an Environmental Report Supplement to comply with 10 CFR Part 51 pursuant to §54.23

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Project Overview

SNC Process -

– Integrated Plant Assessment - Scoping

- SNC developed a comprehensive list of systems and structures and identified functions for each item on the list
- Each function was evaluated against the eight scoping criteria in 10 CFR 54.4(a)(1-3)

– Integrated Plant Assessment - Screening

- As an aid to screening the structures and components, evaluation boundaries were produced for each in-scope function
- Structures and components within the evaluation boundaries were screened to identify those subject to aging management review
- The screening criteria used were those contained in 10 CFR 54.21(a)(1)(i) and (ii)

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Project Overview

SNC Process -

– Integrated Plant Assessment - Aging Management Reviews

- Each structure or component subject to aging management review is included in one or more in-house reviews
- Aging management reviews were performed on a commodity basis (discussion of commodity groups follows)
- Aging effects requiring management were determined systematically for the commodity groups based on materials and environments
- Appropriate program coverage for the structures or components comprising each commodity group was identified or established
- The commodity group/programmatic coverage mapping process is similar to the approach in the GALL report

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Project Overview

SNC Process -

– Integrated Plant Assessment - Aging Management Reviews

- *Plant operating experience was reviewed to validate the determination of aging effects requiring management and as an aid to identify potential enhancement areas*
- *Aging management reviews were summarized into the application (some grouping of AMRs)*
- *The demonstration of adequate aging management is made for each commodity group*

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Project Overview

SNC Process -

– Time Limited Aging Analyses

- SNC created a list of calculations (in-house and A/E) to encompass those with a time-limited nature
- NSSS vendor was contacted separately to review their scope for TLAAs
- An initial screening was performed using criterion 3 - the time-limited nature of the calculation
- The remaining set of calculations was then screened using the remaining 5 criteria
- Both "actives" and "passives" were screened
- Separately, a CLB review was performed to assure a thorough review to identify potential TLAAs

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Project Overview

SNC Process -

– GSI's Addressed in Plant Hatch Application

- *GSI 168 - Environmental Qualification of Electrical Equipment*

  EQ evaluations of electrical equipment are TLAAs. Therefore, this GSI is addressed in Section 4.4 of the Plant Hatch application.

- *GSI 190 - Fatigue Evaluation of Metal Components*

  A number of thermal fatigue evaluations are TLAAs. Therefore, the issue associated with this GSI is addressed in Section 4.2 of the Plant Hatch application.

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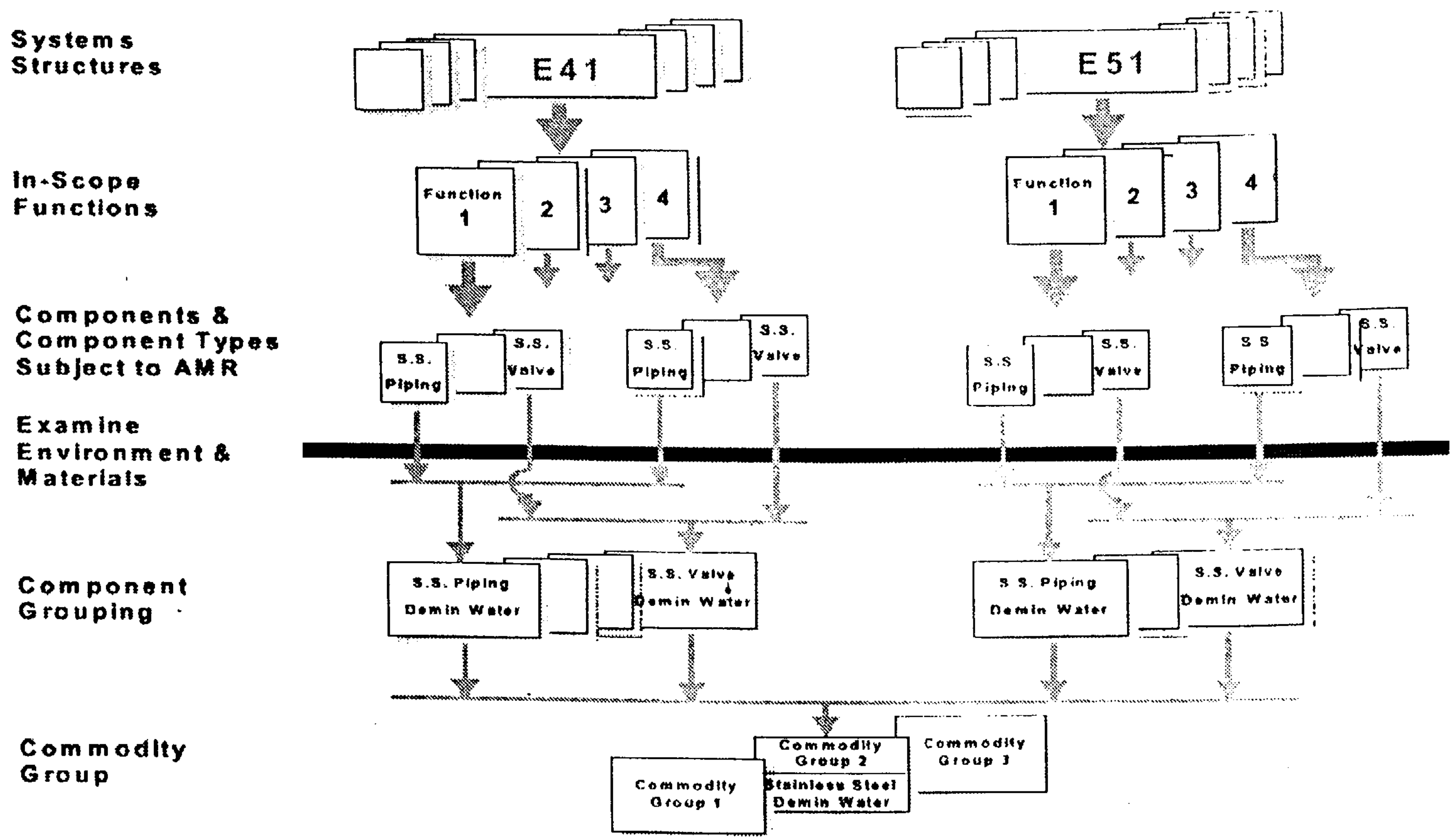
Project Overview

SNC Process -

- *Component groups to commodity groups*



Application Contents

SOUTHERN COMPANY

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Application Contents

- **_License Renewal Application Contents_**
  - **_General Information (section 1)_**
  - **_Structures and Components Requiring an Aging Management Review (section 2)_**
    - _Methodology for scoping and screening_
    - _Identification of structures and components subject to aging management review (scoping/screening results)_
  - **_Aging Management Review Results (section 3)_**
    - _Discussion of process for merging component groups into commodities_
    - _Discussion of aging management review process_

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Application Contents

- ***License Renewal Application Contents (continued)***
  - ***Aging Management Review Results (section 3, contiuned)***
    - *Six-column tables (general overview of aging management reviews results - component types, materials, environments, aging effects requiring management, programs and activities credited)*
  - ***Time-Limited Aging Analyses (section 4)***
    - *Exemptions adddressed (none)*

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Application Contents

- ***License Renewal Application Contents (continued)***
  - ***Final Safety Analysis Report Supplement (appendix A)***
    - *Descriptions of programs and activities for managing aging are contained in appendices A*
  - ***Identification of Aging Effects and Aging Management Review Summaries (appendix C)***
    - *C.1 presents an evaluation of aging effects requiring management*
    - *C.2 presents the summaries of the aging management reviews*
    - *The demonstrations are made in C.2, including the linkage of programs and activities to management of aging effects*

Application Contents

- ***License Renewal Application Contents (continued)***
  - ***Environmental Report Supplement (appendix D)***
  - ***Technical Specification Changes (appendix E)***
    - *Pressure-temperature curve changes to extend operation through 54 effective full-power years*

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Aging Management Reviews

Aging Management Reviews

– ***Systematically identified aging effects requiring management for the following component internal environments:***

- *Reactor grade water*
- *Demineralized water*
- *Suppression Pool water*
- *Spent Fuel Pool water*
- *Borated water*
- *Closed cooling water*
- *Raw water*
- *Fuel oil*
- *Gas*

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Aging Management Reviews

– ***Systematically identified aging effects requiring management for the following component external environments:***
  - *Inside*
  - *Outside*
  - *Buried or Embedded*

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Aging Management Reviews

– **Systematically identified aging effects requiring management for the following structure environments:**
  - *Inside*
  - *Outside*
  - *Buried or Embedded*
  - *Submerged*

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Aging Management Reviews

– **Systematically identified aging effects requiring management for the following electrical component environments:**
  - *High temperature*
  - *Radiation*
  - *Moisture*

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Aging Management Reviews

- Total of 58 AMR summaries are provided in the application and summarize 111 AMRs
  - 45 mechanical
  - 5 electrical
  - 8 civil/structural
- EQ TLAAs manage the vast majority of electrical components subject to aging management review
- Thermal fatigue is managed by combinations of TLAAs and aging management programs

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Aging Management Reviews

– *Mechanical AMR Summaries*

- *45 mechanical AMR summaries*
- *6 mechanical Class 1 AMR summaries are supported by 7 AMRs on-site*
- *28 mechanical non-Class 1 AMR summaries are supported by 91 AMRs on-site*
- *6 mechanical non-Class 1 AMR summaries address fire protection components and are supported by 42 AMRs on-site*
- *3 mechanical AMR summaries address external surfaces*
- *2 AMR summary addresses thermal insulation and jacketing*

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Aging Management Reviews

– **Electrical AMR Summaries**

- 5 electrical AMR summaries for electrical components are provided, supported by 5 in-house AMRs
- only 1 electrical AMR summary addresses an aging effect requiring management outside EQ

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Aging Management Reviews

– *Civil/Structural AMR Summaries*

- *8 civil/structural AMR summaries are provided, supported by 8 in-house AMRs*

AMR Summaries

6 Class 1 Mechanical AMR Summaries

1. Reactor Pressure Vessel
2. Reactor Pressure Vessel Internals
3. Class 1 Carbon Steel Components Within the Reactor Water Environment
4. Class 1 Wrought and Forged Stainless Steel Components Within the Reactor Water Environment
5. Class 1 Cast Austenitic Stainless Steel Components Within the Reactor Water Environment
6. Class 1 Pressure Boundary Bolting

39 non-Class 1 Mechanical AMR Summaries

1. Carbon Steel Components Within the Reactor Water Environment
2. Stainless Steel Components Within the Reactor Water Environment
3. Carbon Steel Components Within the Demineralized Water Environment
4. Stainless Steel Components Within the Demineralized Water Environment
5. Condensate Storage Tanks
6. Carbon Steel Components Within the Suppression Pool Environment
7. Stainless Steel Components Within the Suppression Pool Environment
8. Carbon Steel Components Within the Borated Water Environment
9. Stainless Steel Components Within the Borated Water Environment
10. Carbon Steel Components Within the Closed Cooling Water Environment
11. Stainless Steel Components Within the Closed Cooling Water Environment
12. Copper Alloy Components Within the Closed Cooling Water Environment

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39 non-Class 1 Mechanical AMR Summaries (cont'd.)

13. Carbon Steel Components Within the River Water Environment
14. Stainless Steel Components Within the River Water Environment
15. Copper Alloy Components Within the River Water Environment
16. Gray Cast Iron Components Within the River Water Environment
17. Carbon Steel Components Within the Fuel Oil Environment
18. Stainless Steel Components Within the Fuel Oil Environment
19. Carbon Steel Components Within the Dry Compressed Gas Environment
20. Stainless Steel Components Within the Dry Compressed Gas Environment
21. Copper Alloy Components Within the Dry Compressed Gas Environment
22. Carbon Steel Components Within the Humid or Wetted Gas Environment
23. Stainless Steel Components Within the Humid or Wetted Gas Environment
24. Copper Alloy Components Within the Humid or Wetted Gas Environment

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39 non-Class 1 Mechanical AMR Summaries (cont'd.)

25. Galvanized Carbon Steel Components Within the Humid or Wetted Gas Environment
26. Carbon Steel Bolting Materials
27. Stainless Steel Bolting Materials
28. Residual Heat Removal Heat Exchangers
29. Water Based Fire Suppression Systems
30. Fire Protection Diesel Fuel Oil Supply System
31. Compressed Gas Based Fire Suppression Systems
32. Fire Penetration Seals
33. Cable Tray Fire Barriers
34. Fire Doors

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39 non-Class 1 Mechanical AMR Summaries (cont'd.)

35. Commodity External Surfaces Exposed to an "Inside" Environment
36. Commodity External Surfaces Exposed to an "Outside" Environment
37. Commodity External Surfaces Exposed to a "Buried" or "Embedded" Environment
38. Thermal Insulation
39. Metal Jacketing for Thermal Insulation

8 Structural AMRs

1. Concrete Structures
2. Steel Primary Containment and Internals
3. Steel Structures in Seismic Category I Buildings, the Turbine Building, and Category I Yard Structures
4. Component Supports
5. Spent Fuel Pool Liner, Components, and Racks
6. Aluminum
7. Structural Sealants
8. Tornado Relief Vent Assemblies

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5 Electrical AMR Summaries

1. Phase Bussing
2. Nelson Frames
3. Electrical Splices, Connectors, and Terminal Blocks
4. Insulated Electrical Cable Outside Containment
5. Insulated Electrical Cable - Containment

BWRVIP Reports

- BWRVIP-74 Reactor Pressure Vessel Inspections and Flaw Evaluation Guidelines
- BWRVIP-27 SLC/Core Plate delta-P Inspection and Flaw Evaluation Guidelines
- BWRVIP-38 Shroud Support Inspection and Flaw Evaluation Guidelines
- BWRVIP-41 Jet Pump Assembly Inspection and Flaw Evaluation Guidelines
- BWRVIP-48 Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines
- BWRVIP-76 Core Shroud Inspection and Flaw Evaluation Guidelines
- BWRVIP-18 Core Spray Internals and Flaw Evaluation Guidelines
- BWRVIP-26 Top Guide Inspection and Flaw Evaluation Guidelines
- BWRVIP-47 Lower Plenum Inspection and Flaw Evaluation Guidelines

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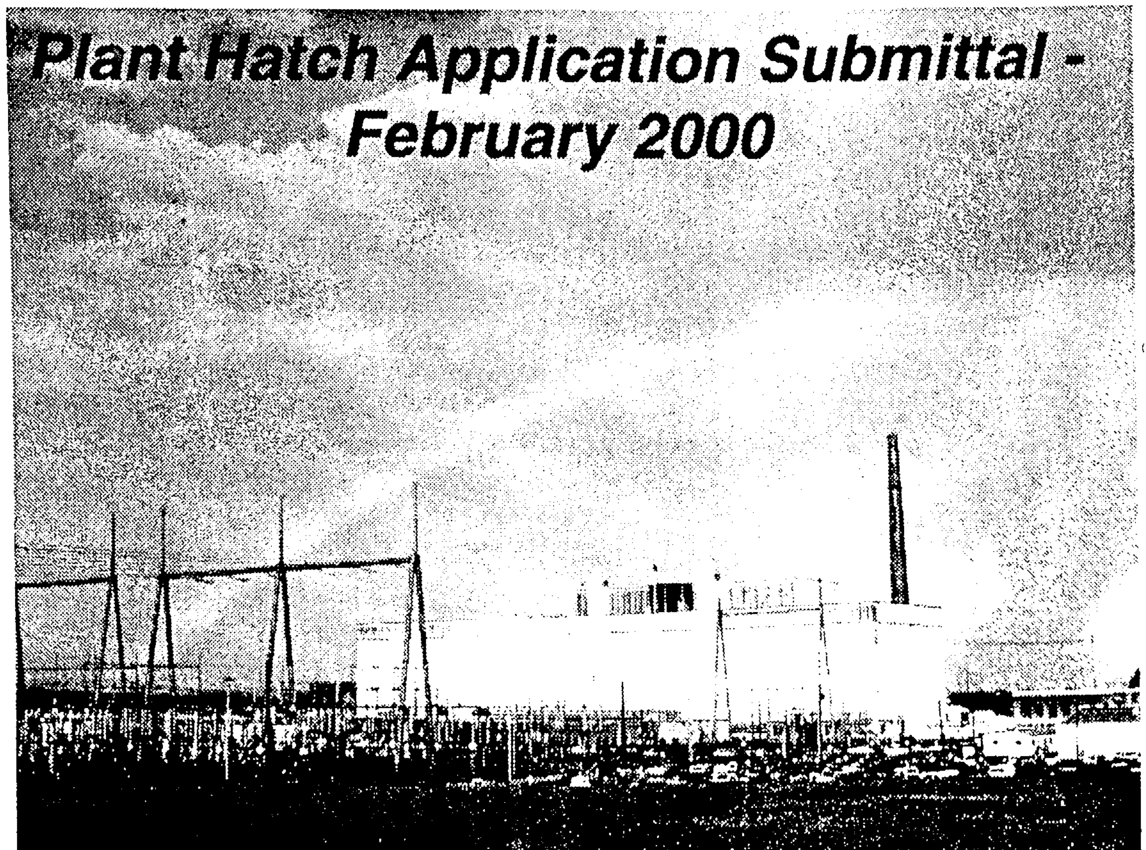
Conclusion



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Applicant's Environmental Report - Operating License Renewal Stage

February 10, 2000

Southern Nuclear Operation Company

SOUTHERN COMPANY
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Project Overview

Our project organization -

Other Major Support Groups:
- Hatch Project Maintenance
- Hatch Project Licensing
- Environmental
- Public Affairs
- Inservice Inspection
- PRA

Hatch Project Management
Lewis Sumner

Technical Services Management
Louis Long

Charter Group

Legal Counsel
Stan Blanton

License Renewal Services Manager
Charles Pierce

Program Management
Expenditure Approval
Report Approval
Legal Issues
NEI Working Group

IPA Process
Computer technology
Application materials
NRC interface
NEI Task Force

Project Manager
Hatch Project
Ray Baker

Engineering Supervisor
Scott Kirk

Team Supervisor
GE Interface

Senior Engineer
Jim Davis

Structural Issues

Environmental Issues
QA issues
Procedure/Maintenance
Deficiency tracking
Communications issues

Senior Engineer
Jeff Mulvehill

Electrical & EQ Issues

Application support
TLAAs & Exemptions
Licensing Submittals

Senior Engineer
Jon Hornbuckle

Mech/Operations Issues

Hatch site interface
Financial analyses
Overall Contractor support

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Environmental Report

Jim Davis
Project Manager

- **SNC Environmental Services**
- **GPC Environmental Lab**
  - Field Survey Oversight
  - Environmental Assessments
  - Environmental Research
  - ER Consultant
- **Tetra Tech NUS**
  - Primary Contractor
  - Environmental Report
  - Environmental Assessments
  - Environmental Research
  - SAMA Analysis
- **Bechtel**
  - SAMA Research
  - SAMA Candidates
  - SAMA Design Description
  - SAMA Cost Estimates

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Environmental Report

- **Format of Environmental Report (ER)**
  - Format of the ER is similar to the Calvert Cliffs ER
  - Section 1.0 - Introduction
  - Section 2.0 - Proposed Action and Alternatives
  - Section 3.0 - Environmental Consequences and Mitigating Actions
  - Section 4.0 - Compliance Status
  - Section 5.0 - References

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Environmental Report

- General Description of the Site
  - Located in Appling County approximately 67 miles southwest of Savannah, Georgia
  - Site consists of 2240 acres
  - Closed loop cooling system
  - Population estimates within 80-km (50-mile radius) obtained using SECPOP90

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Environmental Report

- **Evaluation of the Category 2 Issues**
  - Twenty-one Category 2 issues identified by GEIS
  - Five of the twenty-one requirements not applicable to HNP site
    - Entrainment of fish and shellfish in early life stages (once-through cooling)
    - Impingement of fish and shellfish (once-through cooling)
    - Heat shock (once-through cooling)
    - Groundwater use conflicts (Ranney wells)
    - Groundwater quality degradation (cooling ponds)

- **Sixteen Category 2 issues and Environmental Justice addressed in ER**

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Environmental Report

- *Surface Water Use*
  - *Altamaha river annual flow rate less than 3.15×10^{12} ft^3/year*
  - *HNP withdraws annual avg. of ~ 57 MGD*
  - *HNP returns ~ 25 MGD*
  - *Consumptive loss 0.44 - 3.1 % of river flow*
  - *Impacts small*
  - *GADNR concurred*

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Environmental Report

- **Groundwater Use**
  - HNP pumps more than 100 gallons per minute
  - GADNR permit for four wells at 764 gpm
  - 3 wells installed averaging ~ 126 gpm
  - Nearest appreciable demand is 10 miles south of site
  - Groundwater pump tests determined draw down would not extend to the facility boundary

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Environmental Report

- Terrestrial Resources
  - SNC has no plans to perform major refurbishment activities
  - No impacts to terrestrial resources

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Environmental Report

- *Threatened and Endangered (T&E) Species*
  - *No significant refurbishment activities required for license renewal*
  - *USFWS & NMFS consulted for T&E Federally-listed species*
  - *GADNR consulted for T&E State-listed species*
  - *Environmental field survey performed on the plant site and transmission corridors*
  - *Mussel survey performed in Altamaha*

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Environmental Report

• Threatened and Endangered (T&E) Species

Table 3-2. Listed[1] species known to occur in the vicinity of HNP or in associated rights-of-way.

Common Name	Scientific Name	Federal Status	State Status	Source
Shortnose sturgeon	*Acipenser brevirostrum*	Endangered	Endangered	Reference 5
Eastern indigo snake	*Drymarchon corais couperi*	Threatened	Threatened	Reference 55
Gopher tortoise	*Gopherus polyphemus*	---	Threatened	References 5, 55
American alligator	*Alligator mississippiensis*	Threatened (S/A)	---	References 3, 55
Bald eagle	*Haliaeetus leucocephalus*	Threatened	Endangered	(a)
Wood stork	*Mycteria americana*	Endangered	Endangered	Reference 55
Bachman's sparrow	*Aimophila aestivalis*	---	Rare	Reference 55
Purple honeycomb head	*Balduina atropurpurea*	---	Rare	Reference 65
Cutleaf beardtongue	*Penstemon dissectus*	---	Threatened	Reference 65
Parrot pitcher plant	*Sarracenia psittacina*	---	Threatened	Reference 65
Sandhill golden-aster	*Pityopsis pinifolia*	---	Threatened	(b)
Hairy rattleweed	*Baptisia arachnifera*	---	Endangered	(b)

a. Observed by Georgia Power Company biologists.
b. GNHP
1. Species that USFWS in NMFS has listed or proposed for listing as threatened or endangered; species that GADNR has listed or proposed for listing as rare, threatened, or endangered.

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Environmental Report

- **Air Quality**
  - HNP located in counties classified as attainment for all criteria pollutants
  - Nearest nonattainment area is 140 miles northwest of HNP
  - SNC has no plans to perform major refurbishment activities
  - No impact to air quality

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Environmental Report

- **Microbiological (Thermophilic) Organisms**
  - HNP discharges in a river with annual flow rate less than 3.15×10^{12} ft^3/year
  - GADNR EPD consulted, no known presence in Altamaha
  - HNP discharge would not promote growth
  - No impact to public health
- **Electrical Shock**
  - Description of transmission lines (500 kV and 230 kV lines)
  - 500 kV and 230 kV lines evaluated per NESC for ground clearance
  - Both under NESC 5 mA requirement (ENVIRO)

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Environmental Report

- Housing Impacts
- Public Services, Education
- Public Services, Utilities
- Public Services, Transportation
  - SNC has no plans to perform major refurbishment activities
  - SNC strategic plan does not anticipate an increase workers
  - SNC assumed 60 additional workers for the renewal term for bounding analysis
  - bounding analysis show no socioeconomic impacts

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Environmental Report

- **Offsite Land Use, Refurbishment**
- **Offsite Land Use, License Renewal Term**
  - SNC has no plans to perform major refurbishment activities
  - SNC assumed 60 additional workers for the renewal term for bounding analysis
  - Population-related impacts - small
  - Tax-revenue impacts
    - Currently ~ 70% of Appling County tax base
    - No increase anticipated since no refurbishment improvements planned

Environmental Report

- *Historic and Archaeological Resources*
  - *HNP has no historic or archaeological properties within a 10 mile radius*
  - *No major construction planned for license renewal period*
  - *GADNR Historic Preservation Division concurs*

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Environmental Report

- ***Severe Accident Mitigation Analysis (SAMA)***
  - *Methodology based on "Regulatory Analysis Technical Evaluation Handbook", NUREG/BR-0184, January 1997*
  - *Developed a bounding analysis that reduced risk to "0"*
  - *Analysis demonstrated that any modification greater than $500,000 are not cost justified*
  - *Onsite costs and replacement power were included in analysis*

Environmental Report

Severe Accident Mitigation Analysis (SAMA)

- Candidate Screening Process
  - Initial list of 115 SAMA candidates
  - 16 were already in place at HNP
  - 46 were not applicable or risk significant to BWR design
  - 10 were combined with other similar modifications
  - 43 unique SAMA candidates with potential value remained
  - Preliminary cost estimates were developed for 43
  - These were screened to $500,000
  - 16 remained for more detailed analysis

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Environmental Report

Severe Accident Mitigation Analysis (SAMA)

- *16 Remaining SAMA Candidates*
  - *A more detailed conceptual design and cost analysis was developed*
  - *Level II analysis determined 5 SAMAs were currently adequately covered by plant design and procedures*
  - *One SAMA was found to be greater than $500,000*
  - *The remaining ten were analyzed*
  - *Greatest single SAMA benefit was ~ $2500 with a cost of $100,000 (per unit)*
  - *None of the SAMAs analyzed were cost justified*

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Environmental Report

- **New and Significant Information**
  - Reviewed Category 1 issues to verify that GEIS conclusions remained valid for HNP
  - Met with State and Federal regulatory agencies for input
  - Performed environmental survey for T&E species
  - Performed mussel survey in Altamaha River
  - Environmental Report received environmental, legal and peer reviews
  - Process and procedures in place governed by Environmental Protection Plan to assure new & significant information is adequately addressed.

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Environmental Report

- **Environmental Justice Review**
  - SNC followed guidance in NRR Procedure for Environmental Justice Reviews
  - Evaluation of the Category 2 issues identified no significant environmental impacts
  - Therefore there are no disproportional high and/or adverse impacts on any member of the public

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Environmental Report

- **Alternatives**
  - Feasible Alternatives
    - coal-fired generation
    - gas fired generation
    - imported power
  - Other Alternatives addressed as not feasible
    - wind
    - solar
    - hydropower
    - geothermal
    - wood energy
    - municipal solid waste
    - other biomass-derived fuels
    - oil
    - nuclear power
    - delayed retirement
    - conservation

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Environmental Report

- **Comparison HNP with Alternatives**
  - 1690 MWe Plant Hatch generation
  - 1800 MWe Coal-fired generation
  - 1760 MWe Gas-fired combined-cycle generation
  - 1690 MWe Replacement power

- **Comparison of potential environmental impacts**
  - Land Use
  - Ecology
  - Aesthetics
  - Water Quality
  - Air Quality
  - Solid Waste
  - Human Health
  - Socioeconomics
  - Culture

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Environmental Report

- **Unavoidable Adverse Impacts**
  - No significant adverse impacts associated with the continued operation of HNP were identified
  - No significant refurbishment activities necessary to support continued operation of HNP

- **Irreversible or Irretrievable Resource Commitments**
  - Spent Fuel Assemblies

- **Short-Term Versus Long-Term Productivity**
  - Incremental but small effect on long term air, water and land conditions
  - No long term adverse effects were identified
  - GPC environmental stewardship will enhance productivity

Environmental Report

- *Status of Compliance*
  - *List of HNP permits and compliance status*
  - *SNC personnel responsible for monitoring and ensuring compliance with permits*
  - *SNC has measures in place to ensure environmentally sensitive areas or species of concern are adequately protected*

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Environmental Report

- **Conclusions**
  - Environmental impacts from HNP license renewal are small
  - No unique plant characteristics identified that could affect the environment
  - Federally-listed or State-listed T&E species present on-site or transmission line ROWs will not be impacted
  - No significant historic or archaeological properties located on-site or transmission line ROWs identified
  - No environmental justice issues identified
  - Alternative generation impacts will be greater than license renewal

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From: "Davis, James T." <JTDAVIS@southernco.com>
To: "Jim Wilson (E-mail)" <jhw1@nrc.gov>
Date: Wed, Mar 1, 2000 5:15 PM
Subject: State and Federal Agency Contacts

Jim,

The following list is the principal state and federal agency contacts SNC interfaced with for the development of the Environmental Report. I am having a directory with local governmental and area contacts printed up and I will fed-ex it to you in a few days. It would be helpful if you will provide your preferred mailing address.

Let me know when you schedule your early scouting meeting and I will make the necessary arrangements to support it.

Jim Davis
205.992.7692

Charles Oravetz - Chief, Protected Species Branch
David Bernhart - Fishery Biologist
National Marine Fisheries Service
9721 Executive Center Drive North
St. Petersburg, Florida 33702
(727) 570-5312

Sandra Tucker - Field Supervisor
U.S. Fish and Wildlife Services
Ecological Services Field Office
247 South Milledge Avenue
Athens, Georgia 30605
706-613-9493

Bowers, Mark
US Fish & Wildlife Services
718 Round Oak-Juliette Rd.
Round Oak, GA 31038
(912) 986-3066

Deener, Bert -Fisheries Biologist
Georgia Department of Natural Resources,
Wildlife Resources Division, Fisheries Mgmt Section
108 Darling Avenue
P.O. Box 2089
Waycross, GA 31502-2089
(912) 285-6094

Greg Masson - Assistant Field Supervisor
U.S. Fish and Wildlife Services
Ecological Services Field Office
4270 Norwich Street
Brunswick, Georgia 31520-2521
(912) 265-9336

David J. Waller
Georgia Department of Natural Resources,

R/9

Wildlife Resources Division
2070 U. S. Highway 278, Social Circle, Georgia 30025
770.918.6400

W. Ray Luce - Division Director and Deputy State Historic Preservation Officer
Georgia Department of Natural Resources,
Historic Preservation Division
500 Healey Building
57 Forsyth Street, N. W.
Atlanta, Georgia 30303
404.656.2840

W. M. Winn - Director
Becky J. Blaus - Water Protection Division
Georgia Department of Natural Resources,
Environmental Protection Division, Water Protection Branch
4220 International Parkway
Suite 101
Atlanta, Georgia 30354
404.675.6232

CC: "Baker, Ray D." <RDBAKER@southernco.com>, "Pierce,...

From: James Wilson
To: "JTDAVIS@southernco.com"@GATED.nrcsmtp
Date: Wed, Mar 1, 2000 2:55 PM
Subject: License Renewal Application

Jim:

This is to confirm that we did receive your application yesterday evening. We have distributed the CDs that came with the application - no hard copies yet; they will arrive after the application has been processed by our contractor.

I also wanted to make sure that you had my office and FAX numbers.

(301) 415-1108 - office
(301) 415-1108 - FAX

Thanks for the suggested meeting locations. Because we would like to have two scoping meetings - one from 1:30 to 4:30 in the afternoon and the other from 7:00 to 10:00 in the evening, the school auditoriums would likely not be optimal. The Community Center is booked all day on 5/10, so I have reserved the Federal Savings Bank lodge for that day. I am told that the meeting room is 40 by 60 and will easily accomodate more than 100 persons. It has a kitchen and rest rooms in the back and a large deck overlooking Lake Mayers. It is about 5 miles west of Baxley and about 15 miles from Hatch. Sounds promising.

I would appreciate it if you would furnish me with a set of contacts and their phone numbers from the State resource and permitting agencies that you deal with (Conservation Department, Division of Natural Resources, Fish and Game, State EPA, Historical/Archaelogical Preservation Office, etc.) so that we can begin to liase with them and solicit their participation in the site visit and scoping meeting. It would also be helpful if you could provide me with contacts at the county commission or planning agency.

If you need to discuss environmental matters related to license renewal and are unable to reach me, please contact Cynthia Sochor (e-mail "CSS3@NRC.GOV," telephone 301-415-2462), who will be working with me on this.

Later,

Jim

CC: BXZ, CSS3

From: "Davis, James T." <JTDAVIS@southernco.com>
To: "Jim Wilson (E-mail)" <jhw1@nrc.gov>
Date: Mon, Mar 6, 2000 11:20 AM
Subject: Local Contacts

Jim,

Here is the list of local leaders you requested. If you need other types of local contacts, we can provide them if you identify the types you're interested in.

Jim Davis
205.992.7692

Local Leadership
> Mr. Jeff Baxley
> City Manager
> City of Baxley
> P. O. Box 290
> Baxley, GA 31515
> 912.367.8300
>
> Mayor Steve Rigdon
> City of Baxley
> P.O. Box 290
> Baxley, GA 31515
> 912.367.8300
>
> Mr. Mike Cleland
> Manager
> Appling County
> 100 Oak Street
> Baxley, GA 31513
> 912.367.8100
>
> Mr. Duane Whitley
> Chairman
> Appling County Board of Commissioners
> 100 Oak Street
> Baxley, GA 31513
> 912.367.8100 or
> 912.654.2121
>
> Ms. Allison Flory
> Executive Director
> Baxley/Appling County Chamber of Commerce
> P. O. Box 413
> Baxley, GA 31515
> 912.367.7731
>
> Mr. Dale Atkins
> Director
> Baxley/Appling County Development Authority
> P.O. Box 413
> Baxley, GA 31515
> 912.367.7731

B/11

>
> Sheriff Lewis Parker
> Courthouse Square
> Baxley, GA 31513
> 912.367.8120
>
> Mr. Dane Bruce
> Appling County Emergency Management
> P. O. Box 747
> Baxley, GA 31513
> 912.367.8111
>
> Mayor Ronnie Dixon
> City of Vidalia
> P.O. Box 280
> Vidalia, GA 30474
> 912.537.8718
>
> Mr. Bill Torrence
> City Manager
> City of Vidalia
> P.O. Box 280
> Vidalia, GA 30474
> 912.537.8718
>
> Mayor John Moore
> City of Lyons
> 257 N. State Street
> Lyons, GA 30436
> 912.526.3123
>
> Mr. Rick Hartley
> City Manager
> City of Lyons
> 257 N. State Street
> Lyons, GA 30436
> 912.526.0607
>
> Mr. John Ladson
> Chairman
> Toombs County Development Authority
> 100 Winona Street
> Vidalia, GA 30474
> 912.537-3711
>
> Mr. Bill Mitchell
> President
> Toombs County Chamber of Commerce
> 2805 E. First Street
> Vidalia, GA 30474
> 912.537.4466
>
> Mr. Kendall Palmer
> Chairman
> Toombs County Chamber of Commerce
> 1705 E. First Street

> Vidalia, GA 30474
> 912.537.4466
>
>
> Mr. James Thompson
> Chairman
> Toombs County Board of Commissioners
> P.O. Box 112
> Lyons, GA 30436
> 912.526.3311
>
> Mr. Ronald Widener
> Toombs County Emergency Management
> P.O. Box 487
> Lyons, GA 30436
> 912.526.6424
>
> Mr. Clyde McCall
> Chairman
> Jeff Davis County Board of Commissioners
> P.O. Box 602
> Hazlehurst, GA 31539
> 912.375.4234
>
> Mr. Lonnie Roberts
> County Administrator
> Jeff Davis County Board of Commissioners
> P.O. Box 602
> Hazlehurst, GA 31539
> 912.375.6611
>
>
> Mr. James Dunn
> Jeff Davis County Emergency Management
> 10 Public Safety Dr.
> Hazlehurst, GA 31539
> 912.375.6628
>
> Mayor Roger Williams
> City of Hazlehurst
> P.O. Box 512
> Hazlehurst, GA 31539
> 912.375.6680
>
>
> Mayor Brad Barnard
> City of Reidsville
> P.O. Box 730
> Reidsville, GA 30453
> 912.557.4469
>
>
> Mr. Jerry Burkhalter
> Chairman
> Tattnall County Board of Commissioners
> P.O. Box 25

> Reidsville, GA 30453
> 912.557.4335
>
>
> Mr. Walt Rogers
> Tattnall County Emergency Management
> P.O. Box 905
> Reidsville, GA 30453
> 912.557.6820
>
>
> Mr. Gary McConnell, Director
> GEMA
> P.O. Box 18055
> Atlanta, GA 30316-0055
> 404.624.7000
>
>
> Mr. Pat Cochran
> GEMA
> P.O. Box 18055
> Atlanta, GA 30316-0055
> 404.635.7233
>
> Mr. Jack H. Hutto
> Field Coordinator
> GEMA Area V Field Office
> P.O. Drawer 469
> Alma, GA 31510
> 404.624.7000
>
> Mr. Jim Hardeman
> GA DNR
> 4244 International Pkwy
> Suite 114
> Atlanta, GA 30354
> 404.362.2638
>
> Mr. Ken Davis
> GEMA
> P.O. Box 18055
> Atlanta, GA 30316-0055
> 404.624.7000
>
>

CC: "Baker, Ray D." <RDBAKER@southernco.com>, "Pierce,...

From: "Davis, James T." <JTDAVIS@southernco.com>
To: "Jim Wilson (E-mail)" <jhw1@nrc.gov>, "Mary Ann Pa...
Date: Mon, Mar 13, 2000 2:57 PM
Subject: Word Version of ER Part 1

Jim,

This is the first of several word version sections of the Environmental Report I will be sending you. As I stated in our conversation the application is the official version and there were a few minor changes from the last word version. Most changes were minor and not substantive. The one plant change that we discussed that is different from the application was the addition of 2 additional wells to our well permit for ornamental irrigation. There was no increase in withdrawal limits requested with the additional wells. This plant change will be reflected in our annual update.

<<Appendix-D_Sec1-2.doc>>

CC: "Baker, Ray D." <RDBAKER@southernco.com>, "Pierce,...

B/12

From: "Davis, James T." <JTDAVIS@southernco.com>
To: "Jim Wilson (E-mail)" <jhw1@nrc.gov>, "Mary Ann Pa...
Date: Mon, Mar 13, 2000 3:00 PM
Subject: ER Part 2

<<Appendix-D_Sec3_Part-1.doc>>

<<Appendix-D_Sec4-5.doc>>

CC: "Baker, Ray D." <RDBAKER@southernco.com>, "Pierce,...

From: "Davis, James T." <JTDAVIS@southernco.com>
To: "Jim Wilson (E-mail)" <jhw1@nrc.gov>, "Mary Ann Pa...
Date: Mon, Mar 13, 2000 3:44 PM
Subject: ER Part 3 of 4 again

<<Appendix-D_Sec3_Part-2.doc>>

Just in case, this is without graphics.

From: "Davis, James T." <JTDAVIS@southernco.com>
To: "Jim Wilson (E-mail)" <jhw1@nrc.gov>, "Mary Ann Pa...
Date: Mon, Mar 13, 2000 3:07 PM
Subject: ER Part 4 of 4

<<Appendix-D_Attach-A-B.doc>>

<<Appendix-D_Attach-F.doc>>

CC: "Pierce, Chuck R." <CRPIERCE@southernco.com>, "Bak...

From: "Baker, Ray D." <RDBAKER@southernco.com>
To: "William F. Burton (E-mail)" <wfb@nrc.gov>, "Anand...
Date: Fri, Mar 24, 2000 10:13 AM
Subject: Matrix of Commodities to Programs/Activities

Butch,

As you requested yesterday afternoon, please find attached a WordPerfect document containing a matrix of programs/activities and commodity groups. Please note the following as regards this document:

1. The matrix does not depict where TLAAs are used to disposition certain aging effects for commodity groups (thermal fatigue-related or EQ-related issues).
2. The matrix does not depict those commodity groups for which no aging management is required. These commodity groups largely fall within either the dry gas environment or are electrical commodities.

Also, please note that although reasonable care has been taken to assure the accuracy of this document, it is intended as an administrative tool for your use in making work assignments. The document is not intended to modify the application in any respect.

Please let me know if you have any questions or comments regarding this subject.

Ray Baker
8-992-7367

"Always do right. This will gratify some people and astonish the rest." --
Mark Twain

ATTACHMENT

<<MATRIX_PC_WP.doc>>

R/13

MATRIX - COMMODITIES TO AGING MANAGEMENT PROGRAMS/ACTIVITIES

PROGRAM/ACTIVITY	COMMODITY
A.1.1 REACTOR WATER CHEMISTRY CONTROL	C.2.1.1.1 Reactor Pressure Vessel C.2.1.1.2 Reactor Pressure Vessel Internals C.2.1.1.3 Class 1 Carbon Steel Components Within the Reactor Water Environment C.2.1.1.4 Class 1 Wrought and Forged Stainless Steel Components Within the Reactor Water Environment C.2.1.1.5 Class 1 Cast Austenitic Stainless Steel Components Within the Reactor Water Environment C.2.2.1.1 Non-Class 1 Carbon Steel Components Within the Reactor Water Environment C.2.2.1.2 Non-Class 1 Stainless Steel Components Within the Reactor Water Environment
A.1.2 CLOSED COOLING WATER CHEMISTRY CONTROL	C.2.2.5.1 Non-Class 1 Carbon Steel Components Within the Closed Cooling Water Environment C.2.2.5.2 Non-Class 1 Stainless Steel Components Within the Closed Cooling Water Environment C.2.2.5.3 Non-Class 1 Copper Alloy Components Within the Closed Cooling Water Environment
A.1.3 DIESEL FUEL OIL TESTING	C.2.2.7.1 Non-Class 1 Carbon Steel Components Within the Fuel Oil Environment C.2.2.7.2 Non-Class 1 Stainless Steel Components Within the Fuel Oil Environment C.2.3.2 Fire Protection Diesel Fuel Oil Supply System
A.1.4 PLANT SERVICE WATER AND RHR SERVICE WATER CHEMISTRY CONTROL	C.2.2.6.1 Non-Class 1 Carbon Steel Components Within the River Water Environment C.2.2.6.2 Non-Class 1 Stainless Steel Components Within the River Water Environment C.2.2.6.3 Non-Class 1 Copper Alloys Within the River Water Environment C.2.2.6.4 Non-Class 1 Gray Cast Iron Components Within the River Water Environment C.2.2.11.1 Residual Heat Removal Heat Exchangers
A.1.5 FUEL POOL CHEMISTRY CONTROL	C.2.6.5 Spent Fuel Pool Liner, Components, and Racks C.2.6.6 Aluminum
A.1.6 DEMINERALIZED WATER AND CONDENSATE STORAGE TANK CHEMISTRY CONTROL	C.2.2.2.1 Non-Class 1 Carbon Steel Components Within the Demineralized Water Environment C.2.2.2.2 Non-Class 1 Stainless Steel Components Within the Demineralized Water Environment C.2.2.2.3 Condensate Storage Tanks C.2.2.4.1 Non-Class 1 Carbon Steel Components Within the Borated Water Environment C.2.2.4.2 Non-Class 1 Stainless Steel Components Within the Borated Water Environment
A.1.7 SUPPRESSION POOL CHEMISTRY CONTROL	C.2.2.3.1 Non-Class 1 Carbon Steel Components Within the Suppression Pool Environment C.2.2.3.2 Non-Class 1 Stainless Steel Components Within the Suppression Pool Environment C.2.2.11.1 Residual Heat Removal Heat Exchangers C.2.6.2 Steel Primary Containment and Internals
A.1.8 CORRECTIVE ACTIONS PROGRAM	ALL COMMODITIES

PROGRAM/ACTIVITY	COMMODITY
A.1.9 INSERVICE INSPECTION PROGRAM	C.2.1.1.2 Reactor Pressure Vessel Internals C.2.1.1.3 Class 1 Carbon Steel Components Within the Reactor Water Environment C.2.1.1.4 Class 1 Wrought and Forged Stainless Steel Components Within the Reactor Water Environment C.2.1.1.5 Class 1 Cast Austenitic Stainless Steel Components Within the Reactor Water Environment C.2.1.1.6 Class 1 Pressure Boundary Bolting C.2.2.11.1 Residual Heat Removal Heat Exchangers C.2.4.3 Commodity External Surfaces exposed to a Buried or Embedded Environment C.2.6.2 Steel Primary Containment and Internals
A.1.10 OVERHEAD CRANE AND REFUELING PLATFORM INSPECTIONS	C.2.6.3 Steel Structures in Seismic Category I Buildings, the Turbine Building and Category I Yard Structures
A.1.11 TORQUE ACTIVITIES	C.2.1.1.6 Class 1 Pressure Boundary Bolting C.2.2.10.1 Non-Class 1 Bolting Materials C.2.2.10.2 Non-Class 1 Stainless Steel Bolting Materials
A.1.12 COMPONENT CYCLIC OR TRANSIENT LIMIT PROGRAM	C.2.1.1.1 Reactor Pressure Vessel C.2.1.1.3 Class 1 Carbon Steel Components Within the Reactor Water Environment C.2.1.1.4 Class 1 Wrought and Forged Stainless Steel Components Within the Reactor Water Environment C.2.1.1.5 Class 1 Cast Austenitic Stainless Steel Components Within the Reactor Water Environment C.2.6.2 Steel Primary Containment and Internals
A.1.13 PLANT SERVICE WATER AND RHR SERVICE WATER INSPECTION PROGRAM	C.2.2.6.1 Non-Class 1 Carbon Steel Components Within the River Water Environment C.2.2.6.2 Non-Class 1 Stainless Steel Components Within the River Water Environment C.2.2.6.3 Non-Class 1 Copper Alloys Within the River Water Environment C.2.2.6.4 Non-Class 1 Gray Cast Iron Components Within the River Water Environment C.2.4.3 Commodity External Surfaces exposed to a Buried or Embedded Environment
A.1.14 PRIMARY CONTAINMENT LEAKAGE RATE TESTING PROGRAM	C.2.6.2 Steel Primary Containment and Internals
A.1.15 BOILING WATER REACTOR VESSEL AND INTERNALS PROGRAM	C.2.1.1.2 Reactor Pressure Vessel Internals
A.1.16 WETTED CABLE ACTIVITIES	C.2.5.4 Insulated Electrical Cable Outside Containment
A.1.17 REACTOR PRESSURE VESSEL MONITORING PROGRAM	C.2.1.1.1 Reactor Pressure Vessel

PROGRAM/ACTIVITY	COMMODITY
A.2.1 FIRE PROTECTION ACTIVITIES	C.2.3.1 Water Based Fire Suppression Systems
	C.2.3.2 Fire Protection Diesel Fuel Oil Supply System
	C.2.3.3 Compressed Gas Based Fire Suppression Systems
	C.2.3.4.1 Fire Penetration Seals
	C.2.3.4.2 Cable Tray Fire Barriers
	C.2.3.4.3 Fire Doors
	C.2.4.1 Commodity External Surfaces Exposed to an Inside Environment
	C.2.4.2 Commodity External Surfaces exposed to an Outside Environment
A.2.2 FLOW ACCELERATED CORROSION PROGRAM	C.2.1.1.3 Class 1 Carbon Steel Components Within the Reactor Water Environment
	C.2.2.1.1 Non-Class 1 Carbon Steel Components Within the Reactor Water Environment
A.2.3 PROTECTIVE COATINGS PROGRAM	C.2.2.3.1 Non-Class 1 Carbon Steel Components Within the Suppression Pool Environment
	C.2.2.4.1 Non-Class 1 Carbon Steel Components Within the Borated Water Environment
	C.2.2.10.1 Non-Class 1 Bolting Materials
	C.2.3.1 Water Based Fire Suppression Systems
	C.2.4.1 Commodity External Surfaces Exposed to an Inside Environment
	C.2.4.2 Commodity External Surfaces exposed to an Outside Environment
	C.2.4.3 Commodity External Surfaces exposed to a Buried or Embedded Environment
	C.2.6.1 Concrete Structures
	C.2.6.2 Steel Primary Containment and Internals
	C.2.6.3 Steel Structures in Seismic Category I Buildings, the Turbine Building and Category I Yard Structures
	C.2.6.4 Component Supports
A.2.4 EQUIPMENT AND PIPING INSULATION MONITORING PROGRAM	C.2.4.4.1 Insulation
	C.2.4.4.2 Insulation Jacketing
A.2.5 STRUCTURAL MONITORING PROGRAM	C.2.2.6.1 Non-Class 1 Carbon Steel Components Within the River Water Environment
	C.2.2.6.2 Non-Class 1 Stainless Steel Components Within the River Water Environment
	C.2.2.6.3 Non-Class 1 Copper Alloys Within the River Water Environment
	C.2.2.6.4 Non-Class 1 Gray Cast Iron Components Within the River Water Environment
	C.2.2.11.1 Residual Heat Removal Heat Exchangers
	C.2.6.1 Concrete Structures
	C.2.6.3 Steel Structures in Seismic Category I Buildings, the Turbine Building and Category I Yard Structures
	C.2.6.4 Component Supports
	C.2.6.7 Structural Sealants
	C.2.6.8 Tornado Relief Vent Assemblies

PROGRAM/ACTIVITY	COMMODITY
A.3.1 GALVANIC SUSCEPTIBILITY INSPECTIONS	C.2.1.1.3 Class 1 Carbon Steel Components Within the Reactor Water Environment C.2.2.1.1 Non-Class 1 Carbon Steel Components Within the Reactor Water Environment C.2.2.2.1 Non-Class 1 Carbon Steel Components Within the Demineralized Water Environment C.2.2.3.1 Non-Class 1 Carbon Steel Components Within the Suppression Pool Environment C.2.2.6.1 Non-Class 1 Carbon Steel Components Within the River Water Environment
A.3.2 TREATED WATER SYSTEMS PIPING INSPECTIONS	C.2.1.1.3 Class 1 Carbon Steel Components Within the Reactor Water Environment C.2.1.1.4 Class 1 Wrought and Forged Stainless Steel Components Within the Reactor Water Environment C.2.2.1.1 Non-Class 1 Carbon Steel Components Within the Reactor Water Environment C.2.2.1.2 Non-Class 1 Stainless Steel Components Within the Reactor Water Environment C.2.2.2.1 Non-Class 1 Carbon Steel Components Within the Demineralized Water Environment C.2.2.2.2 Non-Class 1 Stainless Steel Components Within the Demineralized Water Environment C.2.2.3.1 Non-Class 1 Carbon Steel Components Within the Suppression Pool Environment C.2.2.3.2 Non-Class 1 Stainless Steel Components Within the Suppression Pool Environment C.2.2.4.2 Non-Class 1 Stainless Steel Components Within the Borated Water Environment C.2.2.5.1 Non-Class 1 Carbon Steel Components Within the Closed Cooling Water Environment C.2.2.5.2 Non-Class 1 Stainless Steel Components Within the Closed Cooling Water Environment C.2.2.5.3 Non-Class 1 Copper Alloy Components Within the Closed Cooling Water Environment
A.3.3 GAS SYSTEMS COMPONENT INSPECTIONS	C.2.2.9.1 Non-Class 1 Carbon Steel and Cast Iron Components in the Humid or Wetted Gases Environment C.2.2.9.2 Non-Class 1 Stainless Steel Components Containing Humid or Wetted Gases C.2.2.9.3 Non-Class 1 Copper Alloy Components Containing Humid or Wetted Gases C.2.2.9.4 Non-Class 1 Galvanized Carbon Steel and Aluminum Components Containing Humid or Wetted Gases C.2.4.2 Commodity External Surfaces exposed to an Outside Environment C.2.6.7 Structural Sealants
A.3.4 CONDENSATE STORAGE TANK INSPECTION	C.2.2.2.3 Condensate Storage Tanks
A.3.5 PASSIVE COMPONENT INSPECTION ACTIVITIES	C.2.2.9.1 Non-Class 1 Carbon Steel and Cast Iron Components in the Humid or Wetted Gases Environment C.2.2.9.2 Non-Class 1 Stainless Steel Components Containing Humid or Wetted Gases C.2.2.9.4 Non-Class 1 Galvanized Carbon Steel and Aluminum Components Containing Humid or Wetted Gases C.2.4.2 Commodity External Surfaces exposed to an Outside Environment C.2.4.3 Commodity External Surfaces exposed to a Buried or Embedded Environment C.2.6.2 Steel Primary Containment and Internals C.2.6.7 Structural Sealants
A.3.6 RHR HEAT EXCHANGER AUGMENTED INSPECTION AND TESTING PROGRAM	C.2.2.11.1 Residual Heat Removal Heat Exchangers

PROGRAM/ACTIVITY	COMMODITY
A.3.7 TORUS SUBMERGED COMPONENTS INSPECTION PROGRAM	C.2.2.3.1 Non-Class 1 Carbon Steel Components Within the Suppression Pool Environment C.2.2.3.2 Non-Class 1 Stainless Steel Components Within the Suppression Pool Environment

details regarding the site visit will be provided to your staff before the visit. In the interim, should you have any questions regarding the scoping meeting, please contact the NRC Environmental Project Manager, Mr. James H. Wilson, at (301) 415-1108.

Sincerely,

Original Signed By

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation.

Docket No. 50-321 and 50-366

Enclosure: As stated

DISTRIBUTION:

Docket File	CCarpenter	TKenyon	OGC
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OFFICE	RGEB	LA	SC:RGEB	C:RGEB
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OFFICE	C:RLSB	OGC	D:DRIP
NAME	CGrimes	J. Moore/B. Poole	DMatthews
DATE	3/31/2000	4/3/2000	4/4/2000

B/11H

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 4, 2000

Mr. Lewis Sumner, VP Hatch
Southern Nuclear Operating Company
40 Inverness Parkway
Post Office Box 1295
Birmingham, AL 35201

SUBJECT: NOTICE OF INTENT

Dear Mr. Sumner:

On March 1, 2000, the U.S. Nuclear Regulatory Commission (NRC) received your application dated February 29, 2000, to renew the license for Southern Nuclear Operating Company's (SNC) Edwin I. Hatch Nuclear Plant, Units 1 and 2 (Hatch). In support of the application and in accordance with 10 CFR Part 54 and 10 CFR Part 51, SNC submitted an environmental report (ER). The NRC has prepared 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. Enclosed is a copy of the notice of intent, which has been forwarded to the Office of the Federal Register for publication.

Pursuant to 10 CFR 51.28, SNC is invited to participate in the environmental scoping process. As discussed in the enclosed notice, the staff is providing the public with two opportunities to participate in an environmental scoping meeting to be held Wednesday, May 10, 2000. The first session will be held in the afternoon and the second session, covering the same subjects, will be held in the evening. The NRC will present an overview of the environmental review process and will describe how the process will be implemented for the review of the Hatch license renewal application. SNC is invited to participate in both sessions of the meeting by providing a brief description of the proposed action and the ER submitted as part of the application for license renewal. The meetings will be held at the Southeastern Technical Institute in Vidalia, Georgia. A separate NRC meeting notice is being issued with a more detailed agenda. You will be provided a copy of that notice.

Additionally, NRC staff and contractors plan to be on site the week of May 9 - 12, 2000, to discuss the background details of the preparation of the ER with members of your staff, to perform a walkdown of the site and its environs, and to review documentation. Additional

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50–321 AND 50-366

SOUTHERN NUCLEAR OPERATING COMPANY

EDWIN I. HATCH NUCLEAR PLANT

NOTICE OF INTENT TO PREPARE AN ENVIRONMENTAL

IMPACT STATEMENT AND CONDUCT SCOPING PROCESS

Southern Nuclear Operating Company (SNC) has submitted an application for renewal of operating licenses DPR–57and NPF-5 for an additional 20 years of operation at the Edwin I. Hatch Nuclear Plant, Units 1 and 2 (Hatch). Hatch is located in Appling County, Georgia. The application for renewal was submitted by letter dated February 29, 2000, pursuant to 10 CFR Part 54. A notice of receipt of application, including the environmental report (ER), was published in the **Federal Register** on March 10, 2000 (65 FR 13061). A notice of acceptance for docketing of the application for renewal of the facility operating license was published in the **Federal Register** on April 3, 2000 (65 FR 17543). The purpose of this notice is to inform the public that the U.S. Nuclear Regulatory Commission (NRC) will be preparing an environmental impact statement in support of the review of the license renewal application and to provide the public an opportunity to participate in the environmental scoping process as defined in 10 CFR 51.29.

In accordance with 10 CFR 54.23 and 10 CFR 51.53(c), SNC submitted the ER as part of the application. The ER was prepared pursuant to 10 CFR Part 51 and is available for public inspection at the Commission's Public Document Room in the Gelman Building, 2120 L Street, N.W., Washington, DC, 20003-1527. In addition, the Appling County Library, located at

242 East Parker Street, Baxley, GA, 31513, has agreed to make the ER available for public inspection.

This notice advises the public that the NRC intends to gather the information necessary to prepare a plant-specific supplement to the Commission's "Generic Environmental Impact Statement (GEIS) for License Renewal of Nuclear Plants," (NUREG–1437) in support of the review of the application for renewal of the Hatch operating licenses for an additional 20 years. Possible alternatives to the proposed action (license renewal) include no action and reasonable alternative energy sources. 10 CFR 51.95 requires that the NRC prepare a supplement to the GEIS in connection with the renewal of an operating license. This notice is being published in accordance with the National Environmental Policy Act (NEPA) and the NRC's regulations found in 10 CFR Part 51.

The NRC will first conduct a scoping process for the supplement to the GEIS and, as soon as practicable thereafter, will prepare a draft supplement to the GEIS for public comment. Participation in this scoping process by members of the public and local, State, and Federal government agencies is encouraged. The scoping process for the supplement to the GEIS will be used to accomplish the following:

a. Define the proposed action which is to be the subject of the supplement to the GEIS.

b. Determine the scope of the supplement to the GEIS and identify the significant issues to be analyzed in depth.

c. Identify and eliminate from detailed study those issues that are peripheral or that are not significant.

d. Identify any environmental assessments and other environmental impact statements (EISs) that are being or will be prepared that are related to but are not part of the scope of the supplement to the GEIS being considered.

e. Identify other environmental review and consultation requirements related to the proposed action.

f. Indicate the relationship between the timing of the preparation of environmental analyses and the Commission's tentative planning and decision-making schedule.

g. Identify any cooperating agencies and, as appropriate, allocate assignments for preparation and schedules for completing the supplement to the GEIS to the NRC and any cooperating agencies.

h. Describe how the supplement to the GEIS will be prepared, including any contractor assistance to be used.

The NRC invites the following entities to participate in the scoping process:

a. The applicant, Southern Nuclear Operating Company.

b. Any Federal agency that has jurisdiction by law or special expertise with respect to any environmental impact involved, or that is authorized to develop and enforce relevant environmental standards.

c. Affected State and local government agencies, including those authorized to develop and enforce relevant environmental standards.

d. Any affected Indian tribe.

e. Any person who requests or has requested an opportunity to participate in the scoping process.

f. Any person who intends to petition for leave to intervene.

Participation in the scoping process for the supplement to the GEIS does not entitle participants to become parties to the proceeding to which the supplement to the GEIS relates. Notice of opportunity for a hearing regarding the renewal application was the subject of the aforementioned **Federal Register** notice of acceptance for docketing. Matters related to participation in any hearing are outside the scope of matters to be discussed at this public meeting.

In accordance with 10 CFR 51.26, the scoping process for an EIS may include a public scoping meeting to help identify significant issues related to a proposed activity and to determine the scope of issues to be addressed in an EIS. The NRC has decided to hold a public meeting for the Hatch license renewal supplement to the GEIS. The scoping meeting will be held at the small auditorium at the Southeastern Technical Institute 3001 East First Street, Vidalia, Georgia, on Wednesday, May 10, 2000. There will be two sessions to accommodate interested parties. The first session will convene at 1:30 p.m. and will continue until 4:30 p.m. The second session will convene at 7:00 p.m. with a repeat of the overview portions of the meeting and will continue until 10:00 p.m. Both meetings will be transcribed and will include (1) an overview by the NRC staff of the National Environmental Policy Act (NEPA) environmental review process, the proposed scope of the supplement to the GEIS, and the proposed review schedule; (2) an overview by SNC of the proposed action, Hatch license renewal, and the environmental impacts as outlined in the ER; and (3) the opportunity for interested Government agencies, organizations, and individuals to submit comments or suggestions on the environmental issues or the proposed scope of the supplement to the GEIS. Persons may register to attend or present oral comments at the meeting on the NEPA scoping process by contacting Mr. James H. Wilson by telephone at 1 (800) 368–5642, extension 1108, or by Internet to the NRC at *hatcheis@nrc.gov* no later than May 5, 2000. Members of the public may also register to speak at the meeting within 15 minutes of the start of each session. Individual oral comments may be limited by the time available, depending on the number of persons who register. Members of the public who have not registered may also have an opportunity to speak, if time permits. Public comments will be considered in the scoping process for the supplement to the GEIS. If special equipment or accommodations are needed to attend or present information at the public meeting, the need should be brought to Mr. Wilson's attention no later than May 5, 2000, so that the NRC staff can determine whether the request can be accommodated.

Members of the public may send written comments on the environmental scoping process for the supplement to the GEIS to

Chief, Rules and Directives Branch
Division of Administrative Services
Office of Administration
Mailstop T–6 D 59
U.S. Nuclear Regulatory Commission
Washington, DC 20555–0001

Comments may be hand-delivered to the NRC at 11545 Rockville Pike, Rockville, Maryland, between 7:45 a.m. and 4:15 p.m. on Federal workdays. To be considered in the scoping process, written comments should be postmarked by June 9, 2000. Electronic comments may be sent by the Internet to the NRC at *hatcheis@nrc.gov.* Electronic submissions should be sent no later than June 9, 2000, to be considered in the scoping process. Comments will be available electronically and accessible through the NRC's Public Electronic Reading Room (PERR) link http://www.nrc.gov/NRC/ADAMS/index.html at the NRC Homepage.

At the conclusion of the scoping process, the NRC will prepare a concise summary of the determination and conclusions reached, including the significant issues identified, and will send a copy of the summary to each participant in the scoping process. The summary will also be available for inspection through the PERR link. The staff will then prepare and issue for comment the draft supplement to the GEIS, which will be the subject of separate notices and a separate public meeting. Copies will be available for public inspection at the above-mentioned addresses, and one copy per request will be provided free of charge. After receipt and consideration of the comments, the NRC will prepare a final supplement to the GEIS, which will also be available for public inspection.

Information about the proposed action, the supplement to the GEIS, and the scoping process may be obtained from Mr. Wilson at the aforementioned telephone number or e-mail address.

Dated at Rockville, Maryland, this

FOR THE NUCLEAR REGULATORY COMMISSION.

David B. Matthews

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation.

separate public meeting. Copies will be available for public inspection at the above-mentioned addresses, and one copy per request will be provided free of charge. After receipt and consideration of the comments, the NRC will prepare a final supplement to the GEIS, which will also be available for public inspection.

Information about the proposed action, the supplement to the GEIS, and the scoping process may be obtained from Mr. Wilson at the aforementioned telephone number or e-mail address.

Dated at Rockville, Maryland, this 4th day of April 2000.

FOR THE NUCLEAR REGULATORY COMMISSION.

David B. Matthews

David B. Matthews, Director
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation.

From: James Wilson
To: "JTDAVIS@southernco.com"@GATED.nrcsmtp
Date: Fri, Apr 14, 2000 11:13 AM
Subject: Re: Qs on SAMA

Jim:

Here are the questions/clarifications that Bob Palla had for the SAMA review. Bob is out of the office today, but will be back on Monday. If you will give me a couple of time options, I will get with Bob and set up a telecon early next week.

p. F-4: Sequences identified on this and subsequent pages need to be identified. They don't correspond to those identified in the IPE dated December 1992 or the revisions to this IPE, which were noted in the letter dated January 10, 1994. (In the IPE, only 14 sequences were identified; rather than the 15 mentioned in the ER submittal. Release frequencies also differ significantly for the "source term calculation bins" listed in the IPE than for the"sequences" listed in the ER submittal).

p. F-6: The cited CDF cited on this page also puzzles me. As indicated below in Table 1, the Hatch IPE also lists higher numbers that differ for the two units. Where is the reference that describes their LERF results?

Talk with you later, Jim.

CC: BXZ, CSS3

B/15

13/16

GENERIC LICENSE RENEWAL ISSUES

Issue #	Title	Category	Status	Description
98-0001	CREDIT FOR EXISTING PROGRAM	1	RESOLVED	Additional guidance is needed to describe the level of detail needed to take credit for an existing program used to manage the effects of aging. The industry would prefer to limit the descriptions of existing aging management programs, which they believe could potentially expose the CLB, particularly requirements mandated by regulation, to unnecessary review and litigation. Need to address the necessity for a staff review of an existing program under license renewal.
98-0002	DEMONSTRATION DETAILS	3	OPEN	Additional guidance is needed for the content for demonstrating the adequacy of aging management programs pursuant to subpart 54.21(a)(3) and the extent to which the adequacy of aging management can depend on staff review of onsite records rather than docketed.
98-0003	OPERATING EXPERIENCE	3	OPEN	Additional guidance is needed on the scope and details of operating experience that should be addressed to define the applicable aging effects and show that aging management programs are adequate. Discussion should also include operating experience related to new AMPs.
98-0004	SRP - EDITORIAL: USE OF EARLY DETECTION	4	OPEN	Review the SRP-LR 3.6.II.C.1,2,3,4, and determine if "early detection" can be substituted with "detection that provides reasonable assurance..."
98-0005	APPLICABLE AGING EFFECTS	3	OPEN	In order to demonstrate that the effects of aging will be adequately managed for the period of extended operation, as provided in subpart 54.21(a)(3), the application must describe all of the aging effects for the applicable components and structures and determine which of those effects are sufficiently plausible during the period of extended operation to warrant an aging management review. Additional guidance is needed for determining which aging effects are "applicable" during the period of extended operation.
98-0006	GENERIC SAFETY ISSUES	3	MPLEMENTEI	Criteria need to be developed for the scope and level of GSIs to be addressed in a renewal application.
98-0007	RISK-INFORMED LICENSE RENEWAL	3	OPEN	The statements of consideration for the amendment to Part 54 describes how risk insights might be used to assess the importance of safety functions or the adequacy of aging management programs. Additional guidance is needed on the use of risk insights to establish the need for or adequacy of aging management programs.
98-0008	COMPONENT LISTS	3	OPEN	Additional guidance is needed on the appropriate details needed for listing structures and components within the scope of the renewal review, particularly for commodity groups, to satisfy subpart 54.21(a)(1).

Issue #	Title	Category	Status	Description
98-0009	FSAR CONTENT	2	OPEN	Because of the variety in the form and content of existing FSARs, additional guidance is needed on the form, content and timing for the FSAR supplement, and resolution should be consistent with current 50.59 integration effort. Provide guidance for the level of detail needed for existing programs. Keep in mind the need for applicants to identify acceptance criteria and specific parameters being controlled by the amp as well as all other general parameters.
98-0010	TIME-LIMITED AGING ANALYSIS TIMING	3	MPLEMENTEI	Guidance is needed on when time-limited aging analyses (TLAAs) must be submitted for staff review. Many TLAAs will still be valid when the renewal application is submitted and the industry would like to have the option of postponing the analysis updates, including reliance on commitments and license conditions to perform the analysis updates using specified methods after issuance of the renewed license but prior to the expiration of the existing analysis.
98-0011	PASSIVE / ACTIVE DETERMINATION	4	RESOLVED	Part 54 defines "passive" for the purpose of the scope of structures and components subject to review for license renewal. Additional guidance is needed to clarify that definition as it relates to problematic components (for example, fuses and hydrogen recombiners).
98-0012	CONSUMABLES	2	MPLEMENTEI	Some passive components include consumable parts which are replaced during routine maintenance and testing. For example, gaskets and other sealing materials. Additional guidance is needed to establish whether these parts should be included within the scope of the renewal review, to the extent that they are passive in nature and subject to aging effects, or excluded on the basis that they are replaceable components.
98-0013	DEGRADATION INDUCED BY HUMAN ACTIVITIES	4	RESOLVED	Literature and industry experience provide examples of component support degradation by abuse, accidents, and specific or unexpected events. These events potentially cause immediate damage in which case they are not considered aging effects. However, these abuses, accidents, and specific or unexpected events may result in an "event initiated" degradation, which can result in an aging effect. This type of age related degradation is defined as "error-induced aging degradation" by the NRC-approved nuclear power plant aging terminology (reference 2.7). The root cause of failure from error-induced aging degradation is human error not aging. However, the control of error-induced aging degradation is part of aging management.
98-0014	EQ TLAA REQUIREMENTS FOR LRA	2	MPLEMENTEI	Question was raised if the requirements under 10 CFR 50.49 can be used as an aging management program under 10 CFR 54.21(c)(iii) for EQ satisfying the LRA submittal requirements. Review SRP, section 4.4.III.A.3 for any needed changes as a result of this position.
98-0015	ATTRIBUTES OF AN AGING MANAGEMENT PROGRAM	4	OPEN	Related to issue 98-0001 in that a review of the ten attributes of an aging management program as documented in the SRP-LR and NEI 95-10 to develop one list of attributes is used for industry guidance and staff review. The staff also needs to assess the need for all programs to contain all the attributes decided by this review. If possible, this review should identify which attributes are mandatory and which are optional. The SRP-LR and NEI 95-10 need to be revised to reflect this review and development guidance. Keep in mind the potential differences between preventive programs and corrective/repair programs.
98-0016	FUSES, ACTIVE OR PASSIVE	4	RESOLVED	Staff technical position on fuses, active or passive. On 4/10/98, NEI challenged the staff's position that requires a response, which is currently under development. MA5128 - GSI Memo - Fuses

Issue #	Title	Category	Status	Description
98-0017	TRANSFORMERS, ACTIVE OR PASSIVE	4	RESOLVED	Staff technical position on transformers, active or passive.
98-0018	INDICATING LIGHTS (DUAL FILAMENTS), ACTIVE OR PASSIVE	4	RESOLVED	Staff technical position on indicating lights (dual filaments), active or passive.
98-0019	HEAT TRACING, ACTIVE OR PASSIVE	4	RESOLVED	Staff technical position on heat tracing, active or passive.
98-0020	ELECTRICAL HEATERS, ACTIVE OR PASSIVE	4	RESOLVED	Staff technical position on electrical heaters, active or passive.
98-0021	RECOMBINERS, ACTIVE OR PASSIVE	4	RESOLVED	Staff technical position on recombiners, active or passive.
98-0022	DOCKETING INFORMATION	3	OPEN	The staff needs to develop general guidance as to what information will be required of applicants to place on the docket.
98-0023	METHODOLOGY REVIEW TO BE PERFORMED BY WHICH BRANCH	4	OPEN	Need to make a determination as to which branch in NRR will perform the methodology review required under 10 CFR 54.21(a)(2).
98-0024	METHODLOGY REVIEW GUIDANCE	2	MPLEMENTEI	Develop SRP-LR guidance of performing an evaluation of an applicant methodology required under 10 CFR 54.21(a)(2).
98-0026	CUTOFF DATE FOR SUBMITTING A LRA	4	OPEN	Provide guidance in the RG and SRP regarding the CLB cutoff date for preparing a LRA. Because the CLB for an operating plant potentially changes from day-to-day and the applicant must certify that its LRA is accurate at time of submittal, a cutoff date allows an applicant to freeze the documents on which it bases its application.
98-0027	PTS REQUIREMENT UNDER 10 CFR 50.61	3	OPEN	10 CFR 50.21 has a requirement that potentially limits a licensee to submit an application for a renewed license until PTS has been addressed. What is the regulatory requirement as interpreted by OGC.

Issue #	Title	Category	Status	Description
98-0028	FATIGUE OF METAL COMPONENTS	2	OPEN	Develop a staff position on fatigue of metal components for license renewal. This position needs to address a number of specific concerns including environmental fatigue, creep, etc.
98-0029	ENVIRONMENTAL QUALIFICATION OF LOW-VOLTAGE, IN CONTAINMENT CABLES	4	MPLEMENTEI	GSI - 168
98-0030	THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC SS COMPONENTS	2	MPLEMENTEI	The industry wants a distinction between non-significant and potentially significant thermal aging embrittlement effects for CASS.
98-0031	IRRADIATION-ASSISTED SCC OF REACTOR INTERAL COMPONENTS	2	OPEN	The staff considers crack initiation and growth is an applicable aging effect for PWR internals exposed to "high" fluence, but does not define high fluence. The industry wants the staff to define high fluence.
98-0032	STRESS RELAXATION OF PWR INTERNAL COMPONENTS	2	OPEN	EPRI has developed a position on the inspection of internal bolts, pins, and fasteners in PWR reactor vessel internals. The staff needs to develop a position on this specific concern and more generically on PWR reactor vessel internals inspection requirements.
98-0033	PRIMARY WATER SCC OF HIGH-NICKLE ALLOY COMPONENTS	2	OPEN	The industry wants the current ISI and GL 97-01 inspection activities to address SCC of high-nickel alloy components. The staff wants augment inspections because of potential concerns with specific components.
98-0034	SCC OF PWR REACTOR COOLANT SYSTEM COMPONENTS	2	OPEN	The industry wants the current ISI requirements to address SCC in reactor vessel. The staff wants augmented inspections because of the potential of SCC in creviced conditions and stagnant flow regions compounded with the introduction of oxygen into the primary coolant at some facilities.
98-0035	DEGRADATION OF CLASS 1 SMALL BORE PIPING	2	OPEN	Small bore piping is not addressed under current code requirements. However, events experienced with SBP has raised a concern for current operating term as well as for license renewal. Applicability of SBP concerns relative to the current operating term is under review. A position for the renewal term is needed until the current operating term evaluation is complete.
98-0036	NUETRON IRRADIATION EMBRITTLEMENT OF Rx VESSEL AND COMPONETS	2	OPEN	A concern with considering all Rx vessel components that receive the accumulated neutron fluence defined in 10 CFR 50, appendix h, should be considered part of the beltline. The staff disagreed with this statement.

Issue #	Title	Category	Status	Description
98-0037	ULTRASONIC INSPECTION OF PRESSURE VESSEL AND COMPONENTS	2	OPEN	The staff is unwilling to accept ASME Code, Section XI, ultrasonic volumetric ISI without explicit adoption of the mandatory Appendices VII and VIII.
98-0038	VISUAL EXAMINATION OF COMPONENTS AND STRUCTURES	2	OPEN	Adequacy of existing ASME Code, Section XI, visual examination requirements, and the need for augmented inspections to supplement existing examinations.
98-0039	ONE-TIME INSPECTION OF CONCRETE AND STEEL STRUCTURES	3	OPEN	Industry wants to take credit for construction practices and routine visual examinations of exposed concrete and structural steel surfaces for aging management based on no direct mandate by either regulations or by codes and standards. The NRC wants a one-time inspection.
98-0040	FREEZE-THAW DAMAGE IN CONCRETE CONTAINMENT STRUCTURES	3	OPEN	EPRI identifies the CLB design requirements and proposes CLB requirements should be acceptable for LR and a one-time inspection is not needed. The NRC staff needs to develop a position justified by the rule on requiring a one-time inspection. In addtion, include a discussion of this requirement for warm climate plants.
98-0041	ALKALI-AGGREGATE REACTIONS IN CONCRETE CONTAINMENT STRUCTURES	3	OPEN	The SRP indicates that the effects of reaction with aggregates are managed by subsection IWL of Section XI of the ASME code and the requirements of 10 CFR 50.55a. NEI indicates that criteria for determining the potential significance of alkali-aggregate reactions are provided in EPRI TR-107521.
98-0042	DIFFERENTIAL SETTLEMENT IN PWR CONTAINMENT AND CLASS 1 STRUCTURES	2	OPEN	The SRP indicates that acceptable methods for managing the effects of settlement are settlement monitoring during construction and continued monitoring during operation for sites with soft soil and/or significant changes in ground water conditions. Industry indicates that settlement monitoring is not necessary if such a program is not already in place as part of the CLB. NEI provided discussion in EPRI TR-107521.
98-0043	REINFORCEMENT CORROSION IN PWR CONTAINMENTS	3	OPEN	The SRP indicates that acceptable methods for managing the effects of reinforcement corrosion in PWR containments are subsection IWL of Section XI of the ASME code and the requirements of 10 CFR 50.55a. In addition, the management of inaccessible areas of embedded steel should be evaluated. NEI indicates that the criteria for determining the potential significance of reinforcement corrosion are provided in EPRI TR-107521.
98-0044	VOID SWELLING OF REACTOR INTERNALS	2	OPEN	The staff considers void swelling an open license renewal technical issue and the industry does not think it is significant enough to be an issue for LR.
98-0045	SOFTWARE QUALITY CONTROL	4	OPEN	During the review of the BGE FWS report, BGE introduced two pieces of software, "FATIGUEPRO" and "CHECWORKS." This raised the question as to the QC requirements for software used in license renewal.

Issue #	Title	Category	Status	Description
98-0046	INSPECTION OF CONTAINMENT WELDS AND BELLOWS	2	OPEN	The SRP indicates that examination categories e-b and e-f for pressure retaining welds at penetrations and dissimilar metal welds should be performed. In addition, the SRP indicates that augmented examination should be performed for bellows bodies. NEI indicates that the current requirements of 10 CFR 50.55a are adequate for both the current term and the license renewal term.
98-0047	DETECTING AGING EFFECTS IN PLANTS APPLYING FOR LR USING ASME, OM STRDS	3	RESOLVED	RII identified a potential concern relating to adequacy of current component level testing standards based on the ASME O&M Part 21 for heat exchangers and Part 15 and 20 for ECCS systems.
98-0048	ELEVATED TEMPERATURE OF PRESTRESSING TENDONS	3	OPEN	The SRP indicates that elevated temperatures may increase the prestress loss in containment tendons. Thus, an applicant should augment the tendon surveillance program to include additional tendons selected based on their sun exposure or proximity to hot penetrations. NEI indicates that the selection of the 4% pre-stressing tendon sample, in accordance with IWL-2521, is supposed to be random. This should not be solely a license renewal consideration.
98-0049	AGING MANAGMENT OF INACCESSIBLE AREAS OF CONTAINMENT	3	OPEN	Industry concerns with SRP application of ASME IWE/IWL code requirements relative to inaccessible areas. The SRP indicates that an applicant should manage the potential aging effects of containment structures in inaccessible areas when conditions in accessible areas may not indicate the presence of or result in degradation to such inaccessible areas. NEI indicates that this goes well beyond the requirements in 10 CFR 50.55a and is not consistent with the industry reports.
98-0050	IWE/IWL TO INCLUDE BASEMAT	3	DELETED	The SRP addresses the basemat of Mark III containments. Industry questioned whether that basemat is part of the pressure boundary. NEI indicated in a 3/16/99 conference call that the srp is consistent with the industry report and agreed that this issue should be deleted.
98-0051	IWE/IWL, JURISDICTION	3	OPEN	The SRP indicates that examination category f-a is adequate to manage the effects of lockup on certain BWR containment support structures. NEI indicates the f-a should also be cited as adequate to manage corrosion.
98-0052	IWE/IWL, OPERATING EXPERIENCE REQUIREMENTS	3	OPEN	Industry concerns with SRP application of ASME IWE/IWL operating experience documentation.
98-0053	FAILURE DETECTION AS AN ADEQUATE MEANS OF PERFORMANCE MONITORING	2	OPEN	Based on the BGE review of the fire protection system report and the use of flow surveillances as an amp for pressure boundary, a question was raised as the acceptability of failure detection as an amp. Include a discussion on "early detection" as referenced in the SRP. This approach seems inconsistent with the defense in-depth/redundancydesign basis.
98-0054	DEVELOP A LIST OF USIs / GSIs THAT ARE APPLICABLE TO LR.	3	MPLEMENTEI	Develop a list of USIs /GSIs that are applicable to LR.

Issue #	Title	Category	Status	Description
98-0055	CREDIT FOR CONDITION MONITORING AS AN AGING MANAGEMENT PROGRAM	3	DELETED	Additional guidance is needed to describe the level of detail needed to demonstrate the effectiveness of condition monitoring used as an aging management program. The industry would prefer to limit the descriptions of the existing programs, which they believe could potentially expose the CLB, particularly requirements mandated by regulations, to unnecessary review and litigation. The 12/10/98 NEI letter indicates that this issue was created to separate condition monitoring from the existing program issue. NEI requested to delete this issue as a separate item.
98-0056	UPDATING OF FSAR SUPPLEMENT	2	DELETED	Additional guidance is needed to describe the process that will be used to update the FSAR supplement, e.g., can 50.71 be used to update the FSAR supplement for the purpose of license renewal. Use the update guidance to help determine the level of detail as discussed in 98-0009. What level of program change do we want to trigger an FSAR change?
98-0057	CREDITING MAINTENANCE RULE PROGRAM FOR MONITORING STRUCTURAL AGING.	3	OPEN	Need to determine extent to which maintenance rule program can be credited for structural monitoring under license renewal. Also need to consider BTP-PDLR 39.-1 baseline inspection activities to develop SRP guidance. Need to work with the industry to more clearly define their concern.
98-0058	DEFINITION OF BELTLINE REGION	3	OPEN	Resolve differences in the definition of RPB beltline that exist in SRP and 10 CFR 50.61.
98-0059	BOLT CRACKING	2	OPEN	Develop a staff position for aging management of bolting (all bolting within the scope of the AMR) consistent with the rule. Need to evaluate the acceptability of the ASME code requirements for bolting exams and frequency as an acceptable AMP and justify any additions and/or revisions to these requirements. Include in review the wording in the SRP (2.2.iii.a) to ensure the SRP does not refer to bolting as a piece part.
98-0060	INCONSISTANCIES IN SRP	4	OPEN	Review SRP for inconsistencies including the following potential concerns: section 3.4.III.C and - 4.4.II.C, 3.4.III.c and - item 22 of 3.4.II.C, FSAR supplement content, effects of temperature on concrete significance.
98-0061	SRP-EDITORIAL - MORE THAN ONE AMP "SHOULD" "COULD" OR "MAY BE REQUIRED	4	OPEN	Sections 3.0.III.C and 3.0.III.C.2 repeatedly uses "more than one type of AMP should be implemented." Evaluate the use of "should" "could" or "may be required" with respect to the requirements of the rule and make appropriate changes.
98-0062	MONITORING AND TRENDING REQUIREMENT RELATED TO AMR	4	OPEN	Review the monitoring and trending requirements for AMP. NEI has also asked that the staff clarify the need to link monitoring and trending to the degradation mode. Keep in mind an NEI comment that many existing (CLB) programs don't require monitoring/trending when defining the requirements. This may require changes to the SRP for staff review but we also need to be mindful of the need to change NEI 95-10.
98-0063	CORRECTIVE ACTION REQUIREMENTS RELATED TO AMR	4	OPEN	Review corrective action requirements for AMP. Document justification for the need to document any corrective actions for existing programs under the CLB.

Issue #	Title	Category	Status	Description
98-0064	ACCEPTANCE CRITERIA REQUIREMENTS RELATED TO AMR	4	OPEN	Review acceptance criteria requirements for AMP. Discuss the use of quantitative and qualitative acceptance criteria including when either is acceptable and/or required.
98-0065	APPLICANT INSPECTOR QUALIFICATION REQUIREMENTS RELATED TO AMR	4	OPEN	Review any existing or the need for inspector qualification used by licensees in the implementation of AMPs.
98-0066	INSPECTION ACTIVITIES FOR LICENSE RENEWAL, DUPLICATION	4	OPEN	NEI raised a concern relating to inspections of existing program activities that may result in duplicating previous inspections.
98-0067	THE USE OF "EARLY" DETECTION IN REFERENCE TO CRACK DETECTION	4	OPEN	Review the SRP, page 3.2-9, second paragraph, and clarify the use of the term "early" with respect to detection of cracking and intended function.
98-0068	THE USE OF CODE PROGRAM AND CODE EDITIONS	4	OPEN	NEI agreed to clarify the issue by 4/2/99. (Need to document a staff position on the use of code programs and the use of a fixed revisions applicable to license renewal.)
98-0070	HANDLING OF TASKS AND ASSOCIATED SCHEDULES	3	OPEN	Need to develop a staff position on how to handle tasks and schedules outstanding at the time of licensing with respect to enforcing, documenting, and tracking the tasks and schedules. Currently, SRP 3.0.III.d states that these tasks and schedules need to be put in the FSAR.
98-0071	USE OF CONDITION MONITORING AS A MEANS FOR MANAGING AGING FOR A TLAA	2	OPEN	Need to develop a staff position on the use of condition monitoring as a means of managing aging for a TLAA.
98-0072	GUIDANCE FOR DEVELOPING COMMODITY GROUPS	3	OPEN	Need to develop a staff guidance for the development of commodity groups.
98-0073	THE NEED TO INCLUDE EVALUATION BOUNDARIES IN THE SCOPING PROCESS	4	OPEN	The SRP requires reviewers to verify evaluation boundaries are in place and include all structures and components within the scope of the rule. Does the evaluation boundary approach need to be used, if not, correct SRP-LR.
98-0074	SRP - EDITORIAL - THE SRP (3.5.1) REFERS TO 30" PIPING.	4	OPEN	The SRP refers to a specific pipe size. Verify that the specific pipe size used in SRP is accurate across all applicable plants and make any necessary revisions to the SRP.

Issue #	Title	Category	Status	Description
98-0075	HIGH ENERGY LINE BREAKS AS THEY APPLY TO B31.1 PLANTS.	3	OPEN	HELB is not a B31.1 plant design; need to assess applicability to license renewal consistent with the CLB and Part 54.
98-0076	USE OF 10 CFR PART 50, LEVEL OF STAFF REVIEW	4	OPEN	Develop a staff position on the use of 10 CFR part 50 and the level of staff review needed for acceptance. Appendix B is particularly used for corrective actions program and should be discussed in the staff position.
98-0077	TABLES 3.11, SRP-LR, CONSISTENT WITH THE RULE	4	OPEN	NEI is concerned that tables 3.11are not consistent with the rule because of reference to stressors and mechanisms, even though aging effects are also included. Revise the tables or include a discussion in SRP that explains the tables and their tie with the rule.
98-0078	DESIGN CODE REQUIREMENTS FOR CORROSION AS A TLAA	3	OPEN	Provide clarification for considering corrosion as a TLAA under the code in the SRP-LR 4.5.
98-0079	ABNORMAL EVENTS CONTRIBUTION TO AGING EFFECTS.	3	OPEN	Provide a definition of abnormal event in SRP and relate it to event initiated degradation. NEI states that abnormal events need not be postulated for renewal. However, in a recently accepted position on human activities, the staff discuss how an event can impact aging. Include a discussion in SRP-LR 3.0.III.B to clarify this concern, as it relates to PZR cladding as discussed in the B&W PZR SER, and in SRP-LR 3.0.III.B.
98-0080	LEAKAGE FROM BOLTED CONNECTION, AN ABNORMAL EVENTS / ARDM FOR AGING.	3	OPEN	NEI is challenging leakage from a bolted connection as an aging effect. Develop a staff position on leakage from a bolted connection as it relates to aging. Reference 98-0079.
98-0081	SRP-LR USING EVENT INITIATED OCCURRENCES IN DETERMINING AGING EFFECTS	3	OPEN	NEI is challenging two documented operating experiences used incorrectly under 3.2.III.B (1st paragraph) as incidences of aging. Review this section and make necessary corrections. Also make recommendations on how to revise NEI 95-10 as appropriate.
98-0082	HYPOTHETICAL FAILURES/CASCADING SUPPORT SYSTEMS IN SCOPING	2	OPEN	NEI is challenging the use of hypothetical failures. On a 3/16/99 conference call, NEI agreed that this issue addresses "cascading" in scoping. An applicant raised an issue about bounding the scope of systems, structures, and components required under 10 CFR 54.4(a)(3) and not allowing the review to cascade into a review of second-, third-, and fourth-level support systems. In addition, the staff also needs to consider the need to include support systems whose failure could prevent the satisfactory accomplishment of any of the functions required to demonstrate compliance with the commission regulation identified under 10 CFR 54.4(a)(3).
98-0083	SCC IN CARBON STEEL	3	OPEN	Develop a staff position on the potential of SCC in carbon steel.Develop a staff position on the potential of SCC in carbon steel.

Issue #	Title	Category	Status	Description
98-0084	AGING REVIEW OF AIRLOCKS AND EQUIPMENT HATCHES	3	DELETED	The SRP indicates that mechanical wear of airlocks and equipment hatches should be managed in accordance with subsection IWE of Section XI of the ASME Code. NEI is challenging lockup as a possible aging effect for airlocks and related equipment because their position is that related components are active. NEI indicates that this issue should be deleted because it will be addressed by Issue 98-0012 on consumables.
98-0085	REACTOR VESSEL SURVEILLANCE PROGRAM	3	OPEN	In 2/5/99 and 3/16/99 conference calls, NEI indicated that this issue addresses the reactor vessel materials surveillance program for license renewal.
98-0086	INSPECTION OF PZR HEATER BUNDLE PARTIAL PENETRATIONS	3	DELETED	This issue relates to the need for inspection of the pressurizer heater bundle partial penetration welds. NEI indicates that this issue should be deleted because Duke Energy has proposed a program for these inspections for staff review.
98-0087	EVALUATION OF CONTAINMENT TEMPERATURE PROGRAM	3	OPEN	NEI agrees to clarify the issue by 4/2/99. (NEI is challenging the need to perform an evaluation of an applicant's containment temperature program as discussed under SRP-3.4.III.C.4 based on the fact that the CLB should carry forward without reevaluation. Review the SRP and develop a staff position on the discussion presented in the SRP and by NEI.)
98-0088	GENERAL INSPECTION REQUIREMENTS FOR LICENSE RENEWAL.	3	OPEN	NEI has asked, to what extent is inspections required for license renewal in general, to support conclusions that there is no applicable aging effects and are inspections required to validate design assumptions or input such as containment temperature.
98-0089	INTENDED FUNCTION FOR COMMISSION REGULATIONS	4	OPEN	NEI states that section 3.11.II.B of the SRP-LR does not address the intended function of the commission regulations under 54.4(a)(3).
98-0090	CLARIFY "DESIGN BASIS CONDTIONS" AS USED THROUGHOUT SRP.	4	OPEN	Provide clarification of design basis conditions in 3.0.III.B and wherever else it is needed throughout the SRP.
98-0091	IDENTIFYING INTENDED FUNCTIONS FOR COMPLEX STRUCTURES.	4	OPEN	NEI raised a concern that it is difficult to identify component level intended functions for structural components other than "to maintain the structure's intended function."
98-0092	THE SRP LISTS CERTAIN SCs AS NOT PRESENTLY WITHIN THE SCOPE.	4	OPEN	NEI raised a concern that some BWR RVI components are listed as "not presently within the scope." Address concern by determining which components can make a determination on and/or provide a discussion as to what is meant by "presently." Include in discussion the exclusion of the LPCI coupling, and standby liquid control from the scope of LR.

Issue #	Title	Category	Status	Description
98-0093	IASCC AS AN AGING EFFECT FOR CORE SHROUD	3	OPEN	NEI raised a concern that the staff considers IASCC as an applicable aging effect for the core shroud when they provided a justification for not considering it an applicable aging effect for LR.
98-0094	TECHNICAL SPECIFICATION INFORMATION	3	OPEN	Additional guidance is needed for determining what information should be added to the TS or left as a license condition.
98-0095	DEMONSTRATION REQUIREMENTS FOR TLAA	3	OPEN	NEI identified a concern where corrosion allowances originally designed for 60 years still require a measurement to verify that the allowance is good for the period of renewal. Provide additional guidance for demonstration of a TLAA. Use EQ TLAA position as input to staff position.
98-0096	APPLICABILITY OF PIECE-PARTS	3	OPEN	NEI identified a concern with the breakdown of components into piece-parts. Provide a staff position on piece-parts and clarify the application of piece-parts as it exists in SRP 2.2.III.A, page 2.2-3,4.
98-0097	SRP EDITORIAL - USE OF SYSTEM LEVEL VS COMPONENT LEVEL FUNCTIONS.	4	OPEN	Review SRP, page 3.11-3, and the use of Class 1E equipment intended functions that appear to be system level functions but should be component level functions.
98-0098	SRP EDITORIAL - MAY NEED TO INCLUDE < 8" PIPING UNDER 3.6.I.	3	OPEN	The SRP, page 3.6-1, 2nd paragraph, states that only 8" and larger piping are within the scope of LR. This may be incorrect because some smaller piping may have had to be reclassified to Class 1 piping based on application.
98-0099	SRP DEFINITION SECTION	4	OPEN	Add a definition section to the SRP.
98-0100	AGING MANAGEMENT REVIEWS RELATING TO DAMs	3	OPEN	Need a staff position on how the staff will handle the AMR of dams that needs to involve DE, NMSS, and FERC.
98-0101	REVIEW OF GSI-23, 78, 166, AND 173.A	3	MPLEMENTEI	The staff has gained review experience regarding GSI and NUREG-0933 has been updated. The staff has previously provided guidance on review of specific GSI to NEI. The staff needs to evaluate whether the prior staff guidance remains valid.

Issue #	Title	Category	Status	Description
98-0102	SCOPING OF MOTORS/BREAKERS IN STORAGE	3	OPEN	Oconee has some pump motors (for example, high pressure injection pump), electrical breakers, and cabling stored in warehouses onsite to comply with Appendix R of 10 CFR Part 50. The stored equipment would be used in the event of a fire as part of their strategy to comply with Appendix R. Duke identified the cabling to be within scope of license renewal, but considers the stored pump motors and breakers to be outside the scope of renewal. Duke's position is that the renewal rule specifically identifies motors and switchgear to be outside the scope of renewal.
98-0103	NEUTRON EMBRITTLEMENT OF REACTOR VESSEL INTERNALS	2	OPEN	Stainless steel components in the reactor vessel internals are susceptible to neutron embrittlement, which generally results in loss of fracture toughness in the materials. This loss of fracture toughness is a reduction in resistance to crack growth.
99-0104	ACCEPTANCE REVIEW OF A LICENSE RENEWAL APPLICATION	3	OPEN	Section 1.1 of the draft standard review plan for license renewal provides staff review guidance on performing an acceptance review of a license renewal application. This guidance should be enhanced based on lessons learned from the review of the initial license renewal applications.
99-0105	HEAT EXCHANGERS HEAT TRANSFER FUNCTION	3	OPEN	BGE and Duke currently are agreeing with industry position that heat exchangers are active with respect to its heat transfer function. The staff needs to document their position. This is a priority 1 issue because it can impact the license renewal application currently under review by the staff.
99-0106	ULTRASONIC INSPECTION QUALIFICATIONS FOR CONTAINMENTS	3	OPEN	The draft standard review plan for license renewal indicates that Appendices VII and VIII of ASME Section XI should be implemented when ultrasonic examinations are utilized for inspection of containments. NEI commented that the only volumetric examination for containment is a wall thickness measurement which does not meet the purpose of personnel qualifications of Appendices VII and VIII.
99-0107	EROSION OF POROUS CONTAINMENT SUB-FOUNDATION	3	OPEN	NUREG-1611 Indicates that an applicant should evaluate erosion of cement for porous concrete if sub-foundation layers of porous concrete are used in the construction of containment concrete basemat with the presence of underground water. NEI commented that this issue only applies to a few plants, and the current practices that the subject plants have implemented are adequate to manage this aging effect for the renewal term.
99-0108	APPLICATION/SAFETY EVALUATION REPORT FORMAT	4	RESOLVED	There is a need to establish a standard format for license renewal applications using the experience gained from the review of the initial renewal applications and ongoing industry activities. A standard format will assist licensees that may be considering or in the process of developing license renewal applications. The NRC staff has established draft formats for the safety evaluation reports under preparation for the initial renewal applications.

Category 1 = Resolution pending management, Commission or plant-specific action
Category 2 = High Technical Complexity or significant process issue
Category 3 = Medium Technical Complexity or process issue