1 Introduction and General Discussion

1.1 Introduction

This safety evaluation report (SER) documents the Nuclear Regulatory Commission's (NRC's) review of Carolina Power & Light (CP&L) Company's application to renew the operating license for the H.B. Robinson Steam Electric Plant, Unit No. 2, for an additional 20 years. In this SER, HBRSEP, Unit No. 2, is referred to as the Robinson Nuclear Plant (RNP). The NRC's Office of Nuclear Reactor Regulation reviewed the RNP license renewal application (LRA) for compliance with the requirements of Title 10 of the *Code of Federal Regulations*, Part 54 (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

In its June 14, 2002 submittal letter, CP&L requested renewal of the operating license issued under Section 104b of the Atomic Energy Act of 1954, as amended, for RNP (License Number DPR-23) for a period of 20 years beyond the current license expiration date of July 31, 2010. RNP is adjacent to Unit 1 of the H.B. Robinson Steam Electric Plant, a coal-fired steam power plant. RNP is located on Lake Robinson, a man-made lake in Darlington County, South Carolina. RNP is a pressurized light-water, moderated and cooled system. The nuclear power plant incorporates a three-loop closed-cycle, pressurized water, nuclear steam supply system (NSSS) designed by Westinghouse Electric Corporation and designed to generate 2339 MW-thermal, or approximately 769 Mw-electric. Details concerning the plant and the site are found in the Updated Final Safety Analysis Report (UFSAR) for RNP.

The license renewal process proceeds along two tracks: a technical review of safety issues and an environmental review. The requirements for these reviews are stated in NRC regulations 10 CFR Parts 54 and 51, respectively. The safety review for the RNP license renewal is based on the applicant's license renewal application (LRA), RNP UFSAR and on the answers to requests for additional information (RAIs) from the staff. In meetings and docketed correspondence, the applicant has also supplemented its answers to the RAIs. The LRA and all pertinent information and materials, including the UFSAR mentioned above, are available to the public for review at the NRC Public Document Room, 11555 Rockville Pike, Room 1-F21, Rockville, MD, 20852-2738 (301-415-4737/800-3974209). Material related to the LRA is also available through the NRC's website, at www.nrc.gov

This SER summarizes the findings of the staff's safety review of the RNP LRA and delineates the scope of the technical details that the staff considered in its safety evaluation of the proposed operation of Unit 2 for an additional 20 years beyond the terms of the current operating license. The staff reviewed the LRA in accordance with NRC regulations and the guidance in the NRC "Standard Review Plan (SRP) for the Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), dated July 2001.

Sections 2 through 4 of this SER provide the staff's evaluation of the license renewal issues that were considered during the review of the LRA. Section 5 is the report from the Advisory Committee on Reactor Safeguards (ACRS). The conclusions of this report are in Section 6.

Appendix A of this SER is a table that identifies the applicant's commitments associated with the renewal of the operating license. Appendix B is a chronology of the NRC's and the applicant's principal correspondence related to the review of the applications. Appendix C is a list of the NRC staff's principal reviewers and its contractors for this project. Appendix D is a list of the major references used in support of this SER.

In accordance with 10 CFR Part 51, the staff prepared a draft plant-specific supplement to the Generic Environmental Impact Statement (GEIS). This supplement discusses the environmental considerations related to renewing the license for RNP. The draft plant-specific supplement to the GEIS was issued separately as draft Supplement 13 to NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding the H.B. Robinson Steam Electric Plant, Unit 2, in May 2003.

The NRC's project manager for RNP's LRA is Sikhindra K. Mitra. Mr. Mitra may be reached at (301) 415-2783. Correspondence to him should be addressed to License Renewal and Environmental Impacts Program, U.S. Nuclear Regulatory Commission, Mail Stop O-11F1, Washington, D.C. 20555-0001.

1.2 License Renewal Background

Pursuant to the Atomic Energy Act of 1954, as amended, and NRC regulations, licenses for the operation of commercial power reactors are issued for 40 years. These licenses can be renewed for up to 20 additional years. The original 40-year license term was selected on the basis of economic and antitrust considerations, rather than technical limitations. However, some plant equipment may have been designed on the basis of an expected 40-year service life.

In 1982, the NRC anticipated interest in license renewal and held a workshop on the aging of nuclear power plants. This workshop led the NRC to establish a comprehensive program for nuclear plant aging research (NPAR). As a result of this research, a technical review group concluded that many aging phenomena are readily manageable and do not involve technical issues that would preclude extending the life of nuclear power plants.

In 1986, the NRC published a request for comments regarding a policy statement on major policy, technical, and procedural issues related to life extension for nuclear power plants.

In 1991, the NRC published a license renewal rule in 10 CFR Part 54. The NRC participated in an industry-sponsored demonstration program to apply the rule to pilot plants and develop experience to establish implementation guidance. To establish a scope of review for license renewal, the license renewal rule defined age-related degradation unique to license renewal. However, during the demonstration program, the NRC found that many aging mechanisms occur and are managed during the period of the initial license. In addition, the NRC found that the scope of the review did not allow sufficient credit for existing aging management programs (AMPs), particularly programs implemented in accordance with the maintenance rule, 10 CFR 50.65.

As a result, the NRC amended 10 CFR Part 54 in 1995. The amended license renewal rule established a regulatory process that was simpler, more stable, and more predictable than the previous license renewal rule. In particular, 10 CFR Part 54 was clarified to focus on managing the adverse effects of aging rather than on identifying all aging mechanisms. The changes to the license renewal rule were intended to ensure that systems, structures, and components

(SSCs) within the scope of the rule continue to perform their intended functions during the period of extended operation. In addition, the integrated plant assessment (IPA) process was revised and simplified to be consistent with the focus on passive, long-lived structures and components (SCs).

In parallel with these efforts, the NRC pursued a separate rulemaking effort to amend 10 CFR Part 51 to focus the scope of the environmental impact review for license renewal and fulfill, in part, the NRC's responsibilities under the National Environmental Policy Act of 1969 (NEPA).

1.2.1 Safety Review

License renewal requirements for power reactors are based on two principles.

1. The regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety, with the possible exception of the detrimental effects of aging on the functionality of certain system, structures, and components during the period of extended operation and a few other safety issues.

2. The plant-specific licensing basis must be maintained during the renewal term in the same manner, and to the same extent, as during the original licensing term.

In implementing these two principles, 10 CFR 54.4 defines the scope of license renewal as including those plant SSCs (a) that are safety related, (b) whose failure could affect safety-related functions, and (c) that are relied on to demonstrate compliance with the Commission's regulations for fire protection, environmental qualification, pressurized thermal shock (PTS), anticipated transients without scram (ATWS), and station blackout (SBO).

Pursuant to 10 CFR 54.21(a), the applicant must review all SSCs that are within the scope of the rule to identify the SCs that are subject to an aging management review (AMR). SCs that are subject to an AMR are those that perform an intended function without moving parts or without a change in configuration or properties and that are not subject to replacement based on a qualified life or a specified time period. As required by 10 CFR 54.21(a)(3), the applicant must demonstrate that the effects of aging will be managed in such a way that the intended functions of the SCs within the scope of license renewal will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Active equipment, however, is considered to be adequately monitored and maintained by existing programs. In other words, the detrimental effects of aging on active equipment are more readily detectable and will be identified and corrected through routine surveillance, performance indicators, and maintenance. The surveillance and maintenance programs and activities for active equipment and other programs and activities for maintaining plant design and licensing bases are required to continue throughout the period of extended operation.

Pursuant to 10 CFR 54.21(d), each application also is required to include a supplement to the plant's updated final safety analysis report (UFSAR). This UFSAR supplement must contain summary descriptions of the programs and activities for managing the effects of aging.

Another requirement for license renewal is the identification and updating of time-limited aging analyses (TLAAs). During the design phase for a plant, certain assumptions are made about the initial operating term of the plant, and these assumptions are incorporated into design calculations for some of the plant's SSCs. In accordance with 10 CFR 54.21(c)(1), these calculations must be shown to be valid for the period of extended operation or projected to the end of the period of extended operation, or the applicant must demonstrate that the effects of aging of these SSCs will be adequately managed for the period of extended operation.

In July 2001, the NRC issued Regulatory Guide (RG) 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses;" NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Plants" (SRP-LR); and NUREG-1801, "Generic Aging Lessons Learned (GALL) Report." These documents describe methods acceptable to the NRC staff for implementing the license renewal rule and methods used by the NRC staff to evaluate applications for license renewal. The Regulatory Guide endorses a Nuclear Energy Institute (NEI) guideline as an acceptable method of implementing the license renewal rule. The NEI guideline, NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54—The License Renewal Rule," Revision 3, was issued in March 2001. The NRC staff used the regulatory guide along with the SRP-LR to review the RNP LRA.

CP&L utilizes the process defined in NUREG-1801, GALL Report. The purpose of GALL is to provide the staff with a summary of staff-approved aging management programs (AMPs) for the aging of most structures and components that are subject to an aging management review (AMR). If an applicant commits to implementing these staff-approved AMPs, the time, effort, and resources used to review an applicant's LRA will be greatly reduced, thereby improving the efficiency and effectiveness of the license renewal review process. The GALL Report is a compilation of existing programs and activities used by commercial nuclear power plants to manage the aging of structures and components within the scope of license renewal and which are subject to an AMR. The GALL Report summarizes the aging management evaluations, programs, and activities credited for managing aging for most of the structures and components used throughout the industry, and serves as a reference for both applicants and staff reviewers to quickly identify those aging management programs and activities that the staff has determined will provide adequate aging management during the period of extended operation.

1.2.2 Environmental Review

In December 1996, the staff revised the environmental protection regulations in 10 CFR Part 51 to facilitate environmental reviews for license renewal. The staff prepared a "Generic Environmental Impact Statement (GEIS) for License Renewal of Nuclear Plants" (NUREG-1437) to examine the possible environmental impacts of renewing licenses of nuclear power plants. For certain types of environmental impacts, the GEIS establishes generic findings that are applicable to all nuclear power plants. These generic findings are identified as Category 1 issues in 10 CFR Part 51, Subpart A, Appendix B. Pursuant to 10 CFR 51.53(c)(3)(i), an applicant for license renewal may incorporate these generic findings in its environmental report. Environmental impacts that must be analyzed on a plant-specific basis for license renewal are identified as Category 2 issues in 10 CFR Part 51, Subpart A, Appendix B. Such analyses must be included in an environmental report in accordance with 10 CFR 51.53(c)(3)(ii).

In accordance with the NEPA and the requirements of 10 CFR Part 51, the NRC performs a plant-specific review of the environmental impacts of license renewal, including whether there is new and significant information that was not considered in the GEIS. One public meeting was held, near RNP on September 25, 2002, as part of the NRC's scoping process to identify environmental issues specific to the plant. The results of the environmental reviews and preliminary recommendations on the license renewal actions were documented in the NRC draft plant-specific Supplement 13 to the GEIS, which was issued on May 2003, for RNP.

During the 75-day comment period for the draft plant-specific Supplement 13 to the GEIS, an additional public meeting was held, near RNP on June 25, 2003. At this meeting, the staff described the environmental reviews and answered questions from members of the public. After consideration of the comments on the draft, the NRC will prepare and publish a separate final plant specific supplement to the GEIS.

1.3 Summary of Principal Review Matters

The requirements for renewing operating licenses for nuclear power plants are described in 10 CFR Part 54. The staff performed its technical review of the RNP LRAs in accordance with Commission guidance and the requirements of 10 CFR 54.4, 54.19, 54.21, 54.22, 54.23, and 54.25. The standards for renewing a license are set forth in 10 CFR 54.29.

In 10 CFR 54.4, the Commission provides the scoping requirements of the license renewal rule. The applicant submitted this information in Section 2 of its June 14, 2002, applications. The staff reviewed this information and found that the applicant submitted the information required by 10 CFR 54.4.

In 10 CFR 54.19(a), the Commission requires applicants for license renewal to submit general information. The applicant submitted this general information in Enclosure 1 to its letter of June 14, 2002, forwarding its applications for renewed operating licenses for H.B. Robinson Steam Electric Plant Unit 2. The staff reviewed Enclosure 1 and found that the applicant submitted the information required by 10 CFR 54.19(a).

In 10 CFR 54.19(b), the Commission requires that each LRA include "conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license." Regarding the standard indemnity agreement, the applicant states the following in the LRA.

> The current indemnity agreement for H.B. Robinson Steam Electric Plant Unit 2 states in Article VII that the agreement shall terminate at the time of expiration of that license specified in Item 3 of the Attachment to the agreement. Item 3 of the Agreement to the indemnity agreement, as revised by Amendment No. 1, lists H.B. Robinson Operating License DPR-23. CP&L request that conforming changes be made to the indemnity agreement, and/or the Attachment to that agreement, specifying the extension of agreement until the expiration date of the renewed H.B. Robinson operating license as sought in this application. In addition, should the license number be changed upon issuance of the renewed license, CP&L requests that conforming changes be made to the Attachment and any other sections of the indemnity agreement as appropriate.

The staff will use the original license numbers for the renewed licenses. Therefore, there is no need to make conforming changes to the indemnity agreement, and the requirements of 10 CFR 54.19(b) have been met.

In 10 CFR 54.21, the Commission requires that each application for a renewed license for a nuclear facility contain the following information: (a) an IPA, (b) current licensing basis changes made during the NRC review of the application, (c) evaluations of time-limited aging analyses (TLAAs), and (d) an updated final safety analysis report (UFSAR) supplement. On June 17, 2002 the applicant submitted the information required by 10 CFR 54.21(a) and (c) in H.B. Robinson License Renewal Application Exhibit A, "Application for Renewed Operating License, H.B. Robinson Steam Electric Plant Unit 2 LRA Exhibit A, "Application for Renewed Operating License, H.B. Robinson Steam Electric Plant Unit 2." The applicant submitted the information to address the license renewal requirements of 10 CFR 54.21(d) in the UFSAR supplements in Appendix A of the LRA.

In 10 CFR 54.22, the Commission states the requirements regarding technical specifications. In the LRA, Appendix D, the applicant states that no technical specification changes are necessary to manage the effects of aging during the period of extended operation.

The staff evaluated the technical information required by 10 CFR 54.4, 54.21, and 54.22 in accordance with the NRC's regulations and the guidance in the draft SRP. The staff's evaluation of this information is documented in Sections 2, 3, and 4 of this SER.

The staff's evaluation of the environmental information required by 10 CFR 54.23 will be documented in the final plant-specific supplement to the GEIS (NUREG-1437, Supplement 13, dated December 2003).

1.3.1 Westinghouse Topical Reports

In the LRA the applicant referenced certain Westinghouse Commercial Atomic Power (WCAP) reports. In accordance with 10 CFR 54.17(e), the applicant referenced the following WCAP reports in the LRA.

> WCAP-10322, Revision No.1, "Stress Report of 312 Standard Reactor Core Support Structures and Internal Structures Structural and Fatigue Analysis," October, 1984.
>
> WCAP-12962, Supplement 1, "Structural Evaluation of the H.B. Robinson Unit 2 and Shearon Harris Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," October 1995.
>
> WCAP-13587, Revision No. 1, "Reactor Vessel Upper Shelf Energy Bounding Evaluation for Westinghouse Pressurized Water Reactors, September 1993.
>
> WCAP-14209, "Evaluation of the Effects of Insurge/Outsurge Transients of the Integrity of the Pressurizer at H.B. Robinson Unit 2," October 28, 1994.
>
> WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," March 2000.

WCAP-15363, Revision No. 1, “A Demonstration of Applicability of ASME Code Case N-481 to the Primary Loop Pump Casings of H. B. Robinson Unit 2 for the License Renewal Program,” July 2002.

WCAP-15628, “Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the H.B. Robinson Unit 2 Nuclear Power Plant for the License Renewal Program,” July 2001.

The applicant states that in support of license renewal, a new report, WCAP-15363, Revision No. 1, was prepared. WCAP-15363, Revision No. 1, supercedes WCAP-15363, Revision 0, and includes an evaluation of the plant-specific pump casing material properties.

The safety evaluations of the topical reports are intended to be stand-alone documents. An applicant that incorporates the topical reports by reference into an LRA must ensure that the conditions of approval stated in the safety evaluations are met. The staff's evaluation of the applicant’s incorporation of the topical reports into the application is documented in Section 3 of this SER.

1.4 Interim Staff Guidance

The license renewal program is a living program. The NRC staff, industry, and other interested stakeholders gain experience and develop lessons learned with each renewed license. The lessons learned address the NRC’s performance goals of maintaining safety, improving effectiveness and efficiency, reducing regulatory burden, and increasing public confidence. The lessons learned are captured in interim staff guidance (ISG) for use by the staff and interested stakeholders until the improved license renewal guidance documents are revised.

The current set of relevant ISGs that have been issued by the staff and the SER sections where the issues are addressed are provided below:

ISG Issue (**Approved ISG No.)**	**Purpose**	**SER Section**
Station Blackout (SBO) Scoping **(ISG-02)**	The license renewal rule 10 CFR 54.4(a)(3) includes 10 CFR 50.63(a)(1)-SBO. The SBO rule requires that a plant must withstand and recover from an SBO event. The recovery time for offsite power is much faster than that of emergency diesel generators. The offsite power system should be included within the scope of license renewal.	2.5.4 3.6.2.4.3 3.6.2.4.4 3.6.2.4.5
Concrete Aging Management Program **(ISG-03)**	Lessons learned from the GALL Demonstration project indicated that GALL is not clear whether concrete needs any AMPs.	3.5.2.4.1
Fire Protection (FP) System Piping **(ISG-04)**	To clarify staff position for wall thinning of FP piping system in GALL AMPs (XI.M26 and XI.M27). New position is that there is no need to disassemble FP piping, as oxygen can be introduced in the FP piping which can accelerate corrosion. Instead, use nonintrusive method such as volumetric inspection. Testing of sprinkler heads should be performed every 50 years and 10 years after initial service. Eliminated Halon/carbon dioxide system inspections for charging pressure, valve line ups, and automatic mode of operation test from GALL, as the staff considers these test verifications to be operational activities.	2.3.3.15 3.3.2.3.3.2

Identification and Treatment of Electrical Fuse Holder **(ISG-05)**	To include fuse holder AMR and AMP (i.e., same as terminal blocks and other electrical connections). The position includes only fuse holders that are not inside the enclosure of active components (e.g., inside of switchgears and inverters). Operating experience finds that metallic clamps (spring-loaded clips) have a history of age-related failures from aging stressors such as vibration, thermal cycling, mechanical stress, corrosion and chemical contamination. The staff finds that visual inspection of fuse clips is not sufficient to detect the aging effects from fatigue, mechanical stress and vibration.	3.6.2.3.1

1.5 Summary of Open Items

As a result of its review of the LRA for RNP, including additional information submitted to the NRC through April 28, 2003, the staff identified the following issues that remained open at the time this report was prepared. An issue was open if the applicant had not presented a sufficient basis for resolution. Each open item has been assigned a unique identifying number.

Open Item 2.3.1.6-1 (steam generator feedrings)

The staff believes that the steam generator (SG) feedrings should be included in the scope of license renewal (Open Item 2.3.1.6-1). Since this component is completely enclosed by safety-related, pressure-boundary components, it is important to show that failures of this component could not impede certain safety-related functions of the components in which it is contained (10 CFR 54.4(a)(2)).

Open Item 2.3.3.8-1 (exclusion of deepwell pumps, piping, and valves from an AMR)

The staff requested the applicant to provide adequate justification for the exclusion of the deepwell pumps and associated piping from an AMR. The staff found that the applicant has not adequately justified the referred exclusion. The context of Section 10.4.8 of the UFSAR does not link dam failure to any particular set of initiating events, and seismic events and age-related degradation do not encompass all credible causes of dam failure. Dam failure results in loss of the ultimate heat sink and loss of the normal backup supply of feedwater from the service water system through the auxiliary feedwater system. Following dam failure and depletion of the condensate storage tank inventory, failure of the deepwell pumps would cause failure of the safety-related auxiliary feedwater system and prevent the residual heat removal (RHR) necessary to maintain a safe shutdown condition. Therefore, the deepwell pumps and associated piping are within the scope of LR in accordance with 10 CFR 54.4 (a)(2). Therefore, the staff found that the applicant has not adequately justified excluding the deepwell pumps and associated piping and valves from an AMR, and this issue remains as Open Item 2.3.3.8-1.

1.6 Summary of Confirmatory Items

Confirmatory Items are items for which the staff and the applicant have reached a satisfactory resolution, but the resolution has not yet been formally submitted to the staff.

As a result of its review of the LRA for RNP, including additional information submitted to the NRC through April 28, 2003, the staff identified the following issues that remained confirmatory at the time this report was prepared.

Confirmatory Item 2.3.1.3-1 (pressurizer spray head)

The staff believed that the pressurizer spray head should be included in the scope of license renewal (RAI 2.3.1.3-1). Since this component is completely enclosed by safety-related, pressure- boundary components, it is important to show that its failure could not impede certain safety-related functions of the components in which they are contained (10 CFR 54.4(a)(2)). The possibility of a failure in the pressurizer spray head, affecting the functioning of the PORVs or pressurizer safety valves was noted. The applicant surveyed operating experience and concluded that such a failure had not occurred anywhere. The applicant provide supplemental information in support of a revised response to RAI 2.3.1.3-1. Pending the applicant’s formal submittal of this information and the NRC staff’s review of the acceptability of the supplemental information, RAI 2.3.1.3-1 will be considered to be Confirmatory Item 2.3.1.3-1.

Confirmatory Item 2.3.2.5-1 (hydrogen recombiners and supporting components)

The staff considered the applicant's responses to RAIs 2.3.2.5-1, 2.3.2.5-2, and 2.3.2.5-3 to be unacceptable because they are incomplete. Although the responses provided sufficient information to demonstrate that 10 CFR 54.4(a)(1) and (a)(3) did not apply to the hydrogen recombiners and supporting components, they did not adequately demonstrate that these components were not within the scope of license renewal in accordance with 10 CFR 54.4(a)(2). Specifically, although ample time is available to effect hydrogen control, 10 CFR 54.4 does not explicitly permit components required for accident mitigation to be excluded from the scope of license renewal on that basis. In addition, although the response states that sufficient time exists to ensure that all components of the recombiner system are operable before its operation is required, UFSAR Section 6.2.5.2.2 indicates that the majority of the lines associated with this system cannot be repaired due to the high radiation rates present during post-accident conditions. As described further in Section 2.3.2.5.2 of this SER, the applicant has transmitted a revised draft response to these RAIs that would bring within scope the components of the hydrogen recombiner system that are necessary to fulfill the hydrogen control intended function. Pending the applicant’s formal submittal of this information and the

NRC staff’s review of the acceptability of the aging management results for the components that would be added within scope, RAIs 2.3.2.5-1, 2.3.2.5-2, and 2.3.2.5-3 are considered to be Confirmatory Item 2.3.2.5-1.

Confirmatory Item 2.3.3.9-1 (issued with regards to the exclusion from an AMR: refueling water purification pump, piping, and valves necessary for spent fuel pool makeup from the refueling water storage tank)

In discussions regarding the provision of makeup water to the spent fuel pool following loss of cooling, the applicant agreed to include components along the flow path from the refueling water storage tank to the spent fuel pool within the scope of license renewal. The applicant indicated that a revised drawing highlighting the additional components added to the scope of license renewal and a revised list of components (including the purification pump casing, demineralizer vessel, and filter housing) that are subject to an AMR and the associated AMP would be transmitted by letter. This is confirmatory item 2.3.3.9-1

Confirmatory Item 3.0.3.2.2-1 (commitment inspections for the steam generator upper shell-to-transition cone weld)

Confirm that CP&L will commit to performing augmented inspections of the steam generator upper shell-to-transition cone weld during the two 10-year inservice inspection intervals for the extended period of operation for RNP.

Confirmatory Item 3.1.2.1-1, Parts 1 and 2 (issued with regard to the staff’s assessment of AMR Item No. 22 of LRA Table 3.1-1, as evaluated in Section 3.1.2.1 of the SER)

The staff seeks confirmation as to whether or not there is any plant-specific or generic industry experience that supports the conclusion that crack initiation and growth due to stress corrosion cracking (SCC) is an applicable aging effect for carbon steel bolting materials in the RCS. If industry experience does support that crack initiation and growth due to SCC is an applicable aging effect for carbon steel bolting, the applicant should propose an AMP to manage this effect. This is Confirmatory Item 3.1.2.1-1, Part 1.

The applicant’s response to RAI 3.1.2.1-3 states that stress relaxation is not applicable to valve closure bolting in the reactor coolant pressure boundary (i.e., RCPB valve bolting) and “other closure bolting in high pressure and high temperature systems.” However, the applicant’s discussion for AMR 22 to LRA Table 3.1-1 states that the Bolting Integrity Program is applicable to all RCPB bolting except reactor vessel studs for which the Reactor Head Closure Studs Program applies, and that the Bolting Integrity Program relies on the ASME Section XI, Subsection IWB, IWC, and IWD Program to assure that aging effects associated with wear and stress relaxation are managed for RCS Class 1 closure bolting and for Class 2 bolting greater than 2 inches in diameter. The applicant’s discussion of AMR 22 IN LRA Table 3.1-1 did not indicate that the applicant was exempting stress relaxation as an applicable aging effect for the RCPB valve bolting or “other closure bolting in high pressure and high temperature systems.” Therefore, the staff concludes that the applicant’s response to RAI 3.1.2.1-3, as it pertains to the management of stress relaxation in the RCPB valve bolting or “other closure bolting in high pressure and high temperature systems,” contradicts the applicant’s discussion of AMR 22 in LRA Table 3.1-1. The staff requests confirmation that, other than SCC, the aging effects identified in AMR 22 to LRA Table 3.1-1 are still applicable to the RCS bolting within the scope of the commodity group, other than the steam generator primary and secondary manway and handhole bolting. The applicant must explain the contradiction in the RAI response and the information in AMR 22 of LRA Table 3.1-1. This is Confirmatory Item 3.1.2.1-1, Part 2.

Confirmatory Item 3.1.2.1-1, Part 3 (issued with regard to the staff's assessment of AMR Item No. 22 of LRA Table 3.1-1, as evaluated in Section 3.1.2.1 of the SER)

In its response to RAI 3.1.2.1-3, the applicant states that it recognizes that stress relaxation can occur in the SG manway and handhole bolting, at least for the bolting on the secondary side of the SGs, and states that it has a bolting and torque program to determine the closure and torque requirements for reactor cooling system (RCS) closure bolting. However, in its response to RAI 3.1.2.1-3, the applicant did not identify loss of preload as an aging effect and did not identify an aging management program (AMP) to manage the aging effect associated with SG bolting. GALL IV. D.1.1.7 identifies that loss of pre-load due to stress relaxation is an aging effect for the steam generator secondary manway and handhole bolting, and GALL XI.M18, "Bolting Integrity", is the AMP to manage this aging effect. According to 10 CFR 54.21(1), license renewal applicants must perform AMRs and identify all applicable aging effects for passive components within the scope of license renewal. The SG primary and secondary manway and handhole bolts are passive components within the scope of license renewal. The applicant has stated that stress relaxation is an applicable aging effect for the SG secondary manway and handhole bolting; therefore, the applicant is required by 10 CFR 54.21(a)(3) to propose an AMP to manage the aging effect. The staff also requests the applicant to provide technical justification as to why loss of preload stress relaxation does not have to be managed for the primary SG manway bolts in the manner required for the management of the SG secondary side bolting. In subsequent discussions with the NRC staff to resolve this issue, the applicant stated that the RNP bolting integrity program in LRA Section B.3.4 will be applied to the pressure retaining bolting for the primary and secondary side of the steam generators because the RNP bolting integrity program can be relied upon to prevent the loss of preload and that the RNP bolting integrity program will not take exception to the Scope of Program in GALL XI.M18, "Bolting Integrity". The staff evaluates the RNP bolting integrity program in Section 3.0.3 of this SER. The staff finds the applicant's resolution of the issue acceptable because the applicant credits its bolting integrity program to manage loss of preload due to stress relaxation in the SG primary and secondary manway and handhole bolts. However, the applicant needs to submit its resolution under oath and affirmation; therefore, this is Confirmatory Item 3.1.2.1-1, Part 3.

Confirmatory Item 3.1.2.1-2 (issued with regard to the staff's assessment of AMR Item No. 26 of LRA Table 3.1-1, as evaluated in Section 3.1.2.1 of the SER)

In order to provide reasonable assurance that general corrosion is not an applicable aging effect for the Class 1 carbon steel or low steel components in containment air or indoor air environments, the staff seeks confirmation that the Class 1 carbon steel or lower alloy steel components operate at temperatures that are equivalent to or hotter than the ambient temperature for the surrounding containment air or indoor air environments. This is Confirmatory Item 3.1.2.1-2.

Confirmatory Item 3.1.2.1-3, Parts 1 and 2 (issued with regard to the staff's assessment of AMR Item No. 31 of LRA Table 3.1-1, as evaluated in Section 3.1.2.1 of the SER)

The staff seeks confirmation that the reactor vessel (RV) thermal shield is adjacent to the fuel zone region of the RV, receives a neutron fluence greater than 1×10^{17} n/cm2, is within the scope of the commodity group in AMR 31 to LRA Table 3.1-1, and will be managed by the Pressurized Water Reactor Internal Program. This is Confirmatory Item 3.1.2.1-3, Part 1.

The staff seeks confirmation whether or not the RV internal lower support and lower support plate columns are fabricated from cast austenitic stainless steel (CASS) materials and are within the scope of AMRs (i.e., within the scope of AMR Item 8 of LRA Table 3.1-1, AMR Item 33 of LRA Table 3.1-1, and AMR Item 14 of LRA Table 3.1-2). This is Confirmatory Item 3.1.2.1-3, Part 2.

Confirmatory Item 3.1.2.2.4-1 (issued with regard to the staff's assessment of AMR Item No. 6 of LRA Table 3.1-1, as evaluated in Section 3.1.2.2.4 of the SER)

The staff is concerned that the AMPs credited by the applicant for managing crack initiation and growth of small bore Class 1 piping may be used as a precedent for relieving the applicant of performing the required ASME inservice inspection (ISI) examinations for the small bore Class 1 piping welds during the period of extended operation for RNP. Therefore, the staff seeks confirmation that the applicant will continue to perform the ISI examinations of the small bore Class 1 piping that are required by Section XI of the ASME Boiler and Pressure Vessel Code during the period of extended operation for RNP.

Confirmatory Item 3.1.2.2.7-1 (issued with regard to the staff's assessment of AMR Item No. 9 of LRA Table 3.1-1, as evaluated in Section 3.1.2.2.7 of the SER)

The staff seeks confirmation that the welds used to join the SG instrumentation nozzles to the SG shells were fabricated using Alloy 600 weld material (i.e., Alloy 82/182 filler metals). If Alloy 600 weld materials are utilized, the applicant should discuss whether the welds are within the scope of and managed by the Nickel-Alloy Nozzles and Penetrations Program. This is Confirmatory Item 3.1.2.2.7-1.

Confirmatory Item 3.1.2.4.4.3-1 (issued with regard to the staff's assessment of AMR Item No. 10 to LRA Table 3.1-2, as evaluated in Section 3.1.2.4.4.3 of the SER)

The staff seeks confirmation that CP&L is crediting the Nickel-Alloy Nozzles and Penetrations Program as an additional AMP for managing primary water stress corrosion cracking (PWSCC) in the RNP bottom head instrumentation tube nozzles. This is Confirmatory Item 3.1.2.4.4.3-1.

Confirmatory Item 3.1.2.4.5.2-1 (issued with regard to the staff's assessment of AMR Item No. 9 to LRA Table 3.1-2, as evaluated in Section 3.1.2.4.5.2 of the SER)

The staff seeks confirmation that CP&L is crediting the Nickel-Alloy Nozzles and Penetrations Program as an additional AMP for managing PWSCC in the RV core support pads. This is Confirmatory Item 3.1.2.4.5.2-1.

Confirmatory Item 3.3.2.3.3-1 (confirmation that the diesel and motor driven fire pumps are overhauled on a 10-year cycle and this overhaul includes inspection of the bowls)

During the AMR inspection (June 9–13, 2003), the staff reviewed the applicant's replacement frequency for fire pump casings for the Fire Protection Program, see LRA Table 3.3-2, Item 30. The audit noted that there is an error in the application and the fire pumps do not have casings, rather the vertical shaft pumps used at RNP use bowls for the pressure boundary function. Furthermore, the inspection indicated that these bowls are not replaced on a 10 year cycle, rather the pumps are overhauled on a 10-year cycle. Overhaul does not specifically require replacement of the bowls. The applicant explained during a phone call on June 12, 2003, that the frequency of the overhaul of the fire pumps is consistent with OE and that the current Preventive Maintenance Program is effective at ensuring the pumps remain operable during a 10-year service between overhauls. A confirmatory item 3.3.2.3.3-1will be included for the applicant to confirm that the diesel and motor driven fire pumps are overhauled on a 10-year cycle and this overhaul includes inspection of the bowls (i.e., the pressure retaining portion of the pump), and the bowls may or may not be replaced based upon their condition.

Confirmatory Item 3.1.2.4.5.5-1 (Nickel-based alloy incore flux thimbles tubes)

The staff seeks confirmation that the scope of AMR 16 of LRA Table 3.1-2 is for nickel-based alloy incore flux thimbles tubes and not the retractable incore flux thimbles. An inspection-based program should be used in conjunction with the Water Chemistry Program to manage SCC in these components and therefore the staff also seeks confirmation that the applicant will credit both the PWR Vessel Internals Program and the Water Chemistry Program to manage SCC (including PWSCC and/or IASCC) in the nickel-based alloy incore flux thimble tubes. This is Confirmatory Item 3.1.2.4.5.5-1.

Confirmatory Item 3.3.2.4.7-1 (AMP of radioactive equipment drains)

This confirmatory item relates to radioactive equipment drain system (REDS). In RAI 2.3.3.7-2, the staff requested the applicant to clarify which portions of this system are included within the scope of license renewal and subjected to an AMR. In its response dated April 28, 2003, the applicant described the portions of the REDS that are within the scope of license renewal and identified the aging effect of loss of material due to crevice corrosion, pitting corrosion, and microbiologically induced corrosion (MIC). In its response to RAI 2.3.3.7-2, the applicant stated that the identified aging effects do not affect the intended function of the REDS and, therefore, do not require management for the period of extended operation. Based on the information provided in the LRA and the additional information included in the applicant's response to RAI 2.3.3.7-2, the staff requested the applicant to provide additional information to support its conclusion that the identified aging effects do not affect the intended function of the REDS and, therefore, do not require management for the period of extended operation. On June 17, 2003, in a telephone conference, the staff discussed the issue further with the applicant. Subsequent to the telephone conference, by an electronic correspondence dated June 19, 2003, the applicant provided information to support its conclusion on the aging management of REDS. This explanation has been discussed in Section 3.3.2.4.7.2 of this SER. The staff finds that the applicant has provided adequate information to justify that no AMP is required to manage the aging effects of the REDS because the applicant has demonstrated that leaking and blockage of the REDS are unlikely, the potential flow blockage will be identified and corrected timely by the applicant's routine inspection and other activities, and leakage of the REDS would not adversely impact the performance of the SSCs. However, the applicant requested to clarify the applicable aging effects for these REDs components and to incorporate the supporting explanation as discussed above into its response to RAI 2.3.3.7-2. This is Confirmatory Item 3.3.2.4.7-1.

Confirmatory Item 3.3.2.4.17-1 (aging effects for the components in the dedicated shutdown diesel generator)

This confirmatory item relates to the aging effects for the materials and environments associated with the components in the dedicated shutdown diesel generator. In RAI 3.3.17-1, the staff requested the applicant to provide a detailed discussion on the AMR performed for the stainless steel valves, piping, tubing and fittings listed in Table 3.3-2, row numbers 12, 13, and 23, and explain why the AMR results are different among them. In its response, the applicant stated that the air and gas environments in row numbers 12 and 13 include the potential for wetting of stainless steel by untreated water, which is the genesis of the potential aging effects. A detailed explanation of the response has been included in Section 3.3.2.4.17 of this SER. The staff found the referenced explanation appropriate. However, the applicant is requested to provide the above information under oath and affirmation, and this remains as a Confirmatory Item 3.3.2.4.17-1.

Confirmatory Item 3.3.2.4.19-1 (aging effects for the components in the fuel oil system)

This confirmatory item relates to the aging effects for the materials and environments associated with the components in the fuel oil system. In RAI 3.3.17-1, the staff requested the applicant to provide a detailed discussion of the AMR performed for the stainless steel valves, piping, tubing and fittings listed in Table 3.3-2, row numbers 12, 13 and 23, and explain why the AMR results are different among them. The air and gas environments in row numbers 12 and 13 include the potential for wetting of stainless steel by untreated water, which is the genesis of the potential aging effects. In row number 23, the environment is considered a reasonably dry environment which results in no potential aging effects for stainless steel. For the fuel oil system, it has a stainless steel valve and instrumentation tubing, valves, and fittings that are conservatively modeled in a wetted outdoors environment. The fuel oil tank level instrumentation is located outdoors and has components that are near the ground. A detailed explanation of the response has been included in Section 3.3.3.4.19 of this SER. The staff found the referenced explanation appropriate. However, the applicant is requested to provide the above information under oath and affirmation, and this remains as Confirmatory Item 3.3.24.19-1.

Confirmatory Item 3.5-1 (AMP for below-grade reinforced concrete)

In RAI 3.5.1-3, the staff requested the applicant to provide available RNP ground-water chemistry test results including chlorides, sulphate, and pH values and discuss the proposed AMP, as well as past inspection results of below-grade concrete at RNP, since the below-grade reinforced concrete at RNP is exposed to an aggressive environment (low Ph). In RAI 3.5.1-9 the staff stated that it is unclear how the inspection for below-grade containment concrete will be performed by the ASME Section XI, Subsection IWL Program and requested that additional information, such as the locations, depth, and frequency of soil excavation, related to the AMR of below-grade containment concrete be provided. The applicant responded to both RAIs offering commitments that adequately address the staff concerns regarding the aging management of below-grade in-scope concrete structural components at RNP. Because of the slightly acidic RNP ground-water environment, the applicant conservatively assumed existence of an aggressive chemical environment and proposed the plant-specific AMPs (an enhanced ASME, Section XI, Subsection IWL Program for containment and an enhanced Structures Monitoring Program for other Category 1 structures) described in Section 3.5.2.2.1.1 of this SER to manage the aging effects of below-grade concrete. The staff finds RAIs 3.5.1-3 and 3.5.1-9 are fully resolved, pending satisfactory resolution of the Confirmatory Item 3.5-1.

Confirmatory Item 3.6.2.3.1.2-1 (non-EQ insulated cables and connections program)

In LRA Section B.4.6, “Non-EQ Insulated Cables and Connections Program,” the applicant described its AMP to manage aging in Non-EQ insulated cables and connections. The LRA stated that this AMP is consistent with GALL AMPs XI.E1, “Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements” with no deviations. In response to the staff’s concern (RAI B.4.6-2) about excluding non-PVC cables inside and outside containment in adverse localized environment from the sample, the applicant in a letter dated June 13, 2003, stated that the scope of this program includes plant cables and connections of various insulation material types (not just PVC) that may be located in an adverse, localized environment. On the basis of its review, the staff finds that its concern is not resolved. In subsequent discussions with the NRC staff to resolve this issue, the applicant stated that the statement in LRA Section B.4.6 regarding "The sample locations will consider the location of PVC cables inside and outside containment as well as any known adverse localized environments, (PVC was determined to be the limiting insulation material)" will be modified by "The sample locations will consider the location of cables and connections inside and outside containment as well as any known adverse localized environments." The staff finds that the applicant’s resolution of this issue as acceptable because the sample will consider all insulation material types used inside and outside containment as well as any known adverse localized environments. However, the applicant needs to submit its resolution under oath and affirmation; therefore, this is Confirmatory Item 3.6.2.3.1.2-1

Confirmatory Item 3.6.2.3.2.2-1 (AMP for non-EQ electrical cables used in instrumentation circuits (B.4.7)

Operating experience shows that changes in instrument calibration data can be caused by degradation of the circuit cable and are a possible indication of potential cable degradation. The staff finds that the applicant did not address the operating experience in the formal response. In subsequent discussions with the NRC staff to resolve this issue, the applicant stated that this element will be revised to address the operating experience as follows: Industry operating experience indicates that changes in instrument calibration data can be caused by degradation of the circuit cable and are a possible indication of potential cable degradation. This program is for the non-EQ portions of the high range radiation monitoring cabling systems. These cabling systems are located in non-harsh environments and none have experienced age related degradation. The staff finds that the applicant’s resolution of the open item is acceptable because the applicant adequately addressed the operating experience. However, the applicant needs to submit its resolution under oath and affirmation; therefore, this is Confirmatory Item 3.6.2.3.2.2-1.

Confirmatory Item 3.6.2.3.2.2-2 (AMP for neutron flux instrumentation (B.4.8)

To detect aging effects, the cables used in neutron flux instrumentation circuits will be tested at least once every 10 years. Testing may include insulation resistance tests, TDR tests, I/V testing, or other testing judged to be effective in determining cable insulation condition. Following issuance of a renewed operating license for RNP, the initial test will be completed before the end of the initial 40-year license term for Unit 2 (July 31, 2010). The staff finds that this testing is acceptable because the testing will determine cable insulation resistance (potential degradation); however, the staff is concerned about the 10-year testing frequency. This last concern remained as open issue. In subsequent discussions with the NRC staff to resolve this issue, the applicant stated that a review of site operating experience found no age related failures for neutron monitoring cables or connectors. The only industry operating experience identified for these cables was Westinghouse Technical Bulletin 86-01. This Bulletin identified industry concerns with cables used for the source range detector regarding cable degradation due to high operating voltage, radiation, heat, and moisture. Both the source range and intermediate range detector cables inside containment were replaced in 1991 as a result of that bulletin. These cables had operated for 20 years without failure prior to being replaced. The replacement cables were manufactured to Class 1E standards and have remained functional during the last twelve years. The power range cables are the original installed cables and are the same cable type (Amphenol/Essex 21-529) that was originally used in the source range and intermediate range circuits. They have operated for over 32 years without failure, which demonstrates their ability to operate over long periods without a loss of intended function.

In addition, the licensee stated that initial testing of all in-scope neutron monitoring cables will be performed prior to the end of the current license term. This testing will provide a positive means of detecting any significant aging that has occurred since the cables were installed, which in the case of the power range cables will be after 33-40 years of operation. Given the operating experience of these cables and the gradual nature of cable insulation aging, the 10 year testing frequency subsequent to the initial testing provides reasonable assurance that the cables will continue to perform their intended function. The staff finds that the applicant's resolution of the issue is acceptable because the cable insulation degradation is a slow process and RNP operating experience did not identify any cable insulation degradation. Additionally, this 10 year frequency is consistent with NUREG 1801 cable aging management programs frequency. However, the applicant needs to submit its resolution under oath and affirmation; therefore, this is Confirmatory Item 3.6.2.3.2.2-2.

Confirmatory Item 4.2.3-1 (Update of UFSAR Supplement in accordance with the RT_{PTS} and USE values listed in WCAP-15828)

The staff requests confirmation that, at the next update of the UFSAR Supplement for RNP, the applicant will update Sections A.4.2.1 and A.4.2.2 of Appendix A in the LRA to reference the applicability of PTS and USE analyses in WCAP-15828, Revision 0, to the 60-year PTS and USE assessments for the RNP RV beltline materials and will update the corresponding UFSAR

Supplement summary descriptions to reference the RT_{PTS} and USE values listed in the report for the limiting PTS and USE materials in the beltline of the reactor vessel.

Confirmatory Item 4.3.2-1 (auxiliary feedwater fatigue analysis)

In RAI 4.3-7, the staff requested the applicant to provide (1) calculated cumulative utilization factors (CUFs) of the six replacement branch connections, (2) confirmation that no other nonstandard components were used or provide justification of the acceptability for use in safety systems at RNP, and (3) description of the AMPs that will be used to provide assurance that the CUFs for these connections will not exceed the limit of 1.0 for the period of extended operation. In its response by a letter dated June 13, 2003, the applicant stated that there are three 4" to 16" auxiliary feedwater-to-feedwater connections downstream of the motor-driven and the steam-driven AFW pump. The three connections downstream from the steam-driven pumps could not be qualified for the full 40-year design transient set, so a reduced number of design transients was postulated. This resulted in a CUF value of 0.99 for 40-year life. Based upon projections of actual transients to date, the qualified number of transients is not expected to be reached until approximately year 50. The applicant indicated that the number of transients used in the analysis will be tracked by the Fatigue Monitoring Program. The applicant further indicated that the components will be either reanalyzed or replaced prior to exceeding the number of transients tracked by the Fatigue Monitoring Program. The staff finds that the applicant's proposed options provide acceptable plant-specific approaches to address fatigue of the connections between the auxiliary and main feedwater lines for the period of extended operation in accordance with 10 CFR 54.21(c)(1). However, in accordance with 10 CFR 54.21(d), these options need to be included in the UFSAR Supplement (Confirmatory Item 4.3.2-1).

Confirmatory Item 4.3.2-2 (aging management of surge line for period of extended operation)

In RAI 4.3-10, the staff requested that the applicant provide additional clarification regarding aging management of the surge line during the period of extended operation. The applicant's June 13, 2003, response indicated that fatigue of the surge line will be managed using one or more options. Options include further refinement of the fatigue analyses to maintain the environmentally assisted fatigue (EAF)-adjusted CUF below 1.0, or repair of the affected locations, or replacement of the affected locations, or management of the effects of fatigue through the use of an augmented ISI program reviewed and approved by the NRC.

The applicant commits to provide the NRC with the details of the inspection program prior to the period of extended operation if the last option is selected. As indicated by the applicant, the use of an inspection program to manage fatigue will require prior staff review and approval. The applicant indicated that LRA Section A.3.2.2.2 would be revised to include the applicant's proposed options for managing the surge line fatigue. The staff finds the applicant's proposed options provide acceptable plant-specific approaches to address EAF of the RNP pressurizer surge line for the period of extended operation in accordance with 10 CFR 54.21(c)(1). Revision of the UFSAR Supplement is Confirmatory Item 4.3.2-2.

Confirmatory Item 4.6.3-1 (elimination of containment penetration coolers)

This confirmatory item relates to RAI 4.6.3-2. The staff requested the applicant to describe how the analysis was performed and submit the analysis results of concrete properties at the end of 252 cycles. The staff requested the applicant to clarify whether the conclusion of 252 cycles was obtained from its operating experience. During a teleconference call on June 10, 2003, the applicant stated it had found an analysis result indicating that the temperature in concrete around the containment penetration would always remain below 200 ^{o}F. Therefore, the applicant is withdrawing this TLAA item and will submit a new writeup to indicate the withdrawal. Since the applicant's analysis results indicate that the concrete temperature around the containment penetration will always remain below 200 ^{o}F with the elimination of containment penetration coolers, the applicant informed the staff in the teleconference that it had withdrawn this TLAA issue and would submit its new writeup accordingly (Confirmatory Item 4.6.3-1). The staff finds the applicant's approach acceptable.

Confirmatory Item 4.6.4-1 (issued with regard to the staff's assessment of LRA Section B.4.6.4, Aging of Boraflex, as evaluated in Section 4.6.4.2 of the SER)

By letter dated May 28, 2003, the applicant submitted for staff review a license amendment to change the technical specifications regarding removal of Boraflex monitoring procedures. The staff will need confirmation that the license amendment to remove the requirements to credit the Boraflex panels from the RNP technical specification has been approved and that the Boraflex panels will no longer be needed to maintain the K_{eff} for the geometry of the spent fuel rods stored in the spent fuel pool within acceptable levels. As part of this confirmatory item, the staff will need the applicant to provide a reference regarding the staff's safety evaluation to CP&L approving the license amendment for the Boraflex panels. This confirmatory item also requires the applicant 's statement that it will not be necessary to include a summary description of the Boraflex TLAA in the UFSAR Supplement of the application (i.e., in Appendix A of the LRA). This is Confirmatory Item 4.6.4-1.

Confirmatory Item B.3.11-1 (issued with regard to the staff's assessment of LRA Section B.3.11, Reactor Vessel Surveillance Program, as evaluated in Section 3.1.2.3.4 of the SER)

The withdrawal schedule in WCAP-15805 indicates that in-vessel location for Capsule U was moved sometime within the current life of the plant. Therefore, in a meeting with the applicant on May 21, 2003, the staff requested additional clarifying information regarding the elapsed time when Capsule U was moved in the vessel, what the lead factors were for Capsule U at the different in-vessel locations, and what CP&L's basis was for determining that the projected fluence for Capsule U at its projected time of withdrawal would be indicative of the fluence for the RV shell at 50 effective full-power years (EFPY) (i.e., at the EFPY projected for the end of the extended period of operation for RNP). During the meeting of May 21, 2003, the applicant informed the staff that it would provide the additional information requested by the staff. The applicant submitted the requested information in an E-mail to the staff dated June 9, 2003. The applicant must formally submit the information in the E-mail of June 9, 2003, into the docket for RNP (i.e., into Docket No. 50-261) under "Oath and Affirmation." This is Confirmatory Item B.3.11-1.

Confirmatory Item B.4.1-1 (issued with regard to the staff's assessment of LRA Section B.4.1, Nickel-Alloy Nozzles and Penetrations Program, as evaluated in Section 3.1.2.3.6 of the LRA)

The first paragraph in the UFSAR Supplement summary description for the Nickel-Alloy Nozzles and Penetrations Program is not up to date and needs to be amended to reflect that the applicant's inspection program for the RNP vessel head penetration (VHP) nozzles is based on the requirements in NRC Order No. EA-03-009 (February 11, 2003) and the applicant's response to the order dated March 3, 2003. The applicant must confirm that the UFSAR Supplement summary description for the Nickel-Alloy Nozzles and Penetrations Program (as given in Section A.3.1.28 of Appendix A to the LRA) will be amended to reflect the augmented requirements in NRC Order No. EA-03-009 for the RNP upper reactor vessel head and its VHP nozzles. This is Confirmatory Item B.4.1-1.

Confirmatory Item B.4.2-1 (issued with regard to the staff's assessment of LRA Section B.4.2, Thermal Aging of Cast Austenitic Stainless Steel Program, as evaluated in Section 3.1.2.3.7 of the SER)

The staff seeks confirmation that, although an LBB flaw tolerance evaluation has been performed for the extended period of operation for RNP (as given in WCAP-15628), the applicant will continue to perform those ISI examinations for the primary coolant loop piping, valve, and pump casings that are required by Table IWB-2500-1 of Section XI to the ASME Boiler and Pressure Vessel Code, unless relief has been granted by the NRC under applicable provisions in 10 CFR 50.55a from meeting the staff's ISI requirements of 10 CFR 50.55a(g)(4). If relief has been granted from any of the required ISI examinations for the primary coolant loop piping, valve, or pump casings, the staff seeks confirmation of the applicable NRC staff safety evaluation granting this relief and the specific ISI examination requirements for which relief has been granted. The staff also seeks confirmation that the UFSAR Supplement summary description will be amended to reflect the information in the applicant's response to this confirmatory item. This is Confirmatory Item B.4.2-1.

Confirmatory Item B.4.3-1 (issued with regard to the staff's assessment of LRA Section B.4.3, PWR Vessel Internals Program, as evaluated in Section 3.1.2.3.8 of the SER)

The staff will confirm that the applicant has incorporated the commitment regarding the Nickel-Alloy Nozzles and Penetrations Program into the UFSAR Supplement summary description of Section A.3.1.30 of Appendix A to the LRA when the applicant revises its UFSAR Supplement for this AMP. This is Confirmatory Item B.4.3-1.

1.7 Summary of Proposed License Conditions

As a result of the staff's review of the RNP application for license renewal, including the additional information and clarifications submitted subsequently, the staff identified two proposed license conditions. The first license condition requires the applicant to include the UFSAR Supplement in the next UFSAR update required by 10 CFR 50.71(e) following issuance of the renewed license. The second license condition requires that the future inspection activities identified in the UFSAR Supplement be completed prior to the period of extended operation.