June 30, 2003

MEMORANDUM TO: Samson S. Lee, Section Chief
License Renewal Section
Division of Regulatory Improvement Programs

FROM: Frank Akstulewicz, Chief ***/RA/***
BWR Systems and Nuclear Performance Section
Reactor Systems Branch
Division of Systems Safety and Analysis

SUBJECT: REVIEW OF SCOPE FOR ROBINSON LICENSE RENEWAL APPLICATION (TAC NO. MB5223)

By letter dated June 14, 2002, Carolinas Power & Light Company (CP&L) filed an application for renewal of the Operating License for their H.B. Robinson Steam Electric Plant, Unit No. 2, to extend the term of said Operating License by an additional 20 years, until July 31, 2030.

By letter dated October 23, 2002, CP&L supplemented their license renewal application with information concerning the Interim Staff Guidance pertaining to fire protection system aging management, station blackout, aging management of concrete components, and 10 CFR 54.4(a)(2).

The SRXB staff has reviewed the information, submitted by CP&L, in order to ascertain whether the scope of systems, structures, and components (SSCs), as identified by CP&L, is complete, and in accordance with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1).

The SRXB staff have completed their review of CP&L's license renewal scope, and have provided their results and conclusions in the attached Safety Evaluation (SE). The SRXB staff have concluded that the licensee's identification of affected SSCs would be acceptable, with the addition of the pressurizer spray head and the steam generator feedrings. This completes the SRXB scope of work under TAC NO. MB5223.

Attachment: As Stated

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO THE LICENSE RENEWAL APPLICATION
FOR H.B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2
TAC NO. MB5223

Note: This Safety Evaluation has been written according to the template supplied by the Project Manager. Two Open Items have been identified herein as SM-1 and SM-2. (These Items might be renamed by the Project Manager.) These Open Items are found in the pressurizer and steam generator scope evaluations, and call for the addition of the pressurizer spray head and the steam generator feedrings to the license renewal scope.

2.2 Plant-Level Scoping Results

2.2.1 Summary of Technical Information in the Application

This section addresses the plant-level scoping results for license renewal. 10 CFR 54.21(a)(1) requires the applicant to identify and list structures and components subject to an AMR. These are passive and long-lived SCs that are within the scope of license renewal.

In LRA Table 2.2-1, the applicant provided a list of the plant systems and structures, identifying those that are within the scope of license renewal. The Rule does not require the identification of all plant systems and structures. However, providing such a list allows for a more efficient staff review. On the basis of the design basis events considered in the plant's CLB, and other CLB information relating to non-safety-related systems and structures, and certain regulated events, the applicant identified those plant-level systems and structures within the scope of license renewal, as defined in 10 CFR 54.4(a). To verify that the applicant has properly implemented its methodology, the staff focuses its review on the implementation results to confirm that there is no omission of plant-level systems and structures within the scope of license renewal.

2.2.2 Staff Evaluation

In LRA Section 2.1, the applicant describes its methodology for identifying the SCs that are within the scope of license renewal and subject to an AMR. This methodology typically consists of a review of all plant SSCs to identify those that are within the scope of license renewal in accordance with the requirements of 10 CFR 54.4. From those SSCs that are within the scope of license renewal, an applicant will identify and list those SCs that are passive (i.e., that perform their intended function(s) without moving parts, or without a change in configuration or properties), and are long-lived (i.e., that are not replaced based on a qualified life or specified time period). The staff reviewed the scoping and screening methodology, and provided its evaluation in Section 2.1 of this SER. The applicant documented the implementation of the methodology in LRA Sections 2.3 through 2.5. The staff's review of the applicant's implementation can be found in Sections 2.3 through 2.5 of this SER.

Attachment

To ensure that the scoping and screening methodology described in LRA Section 2.1 was properly implemented, and that the SCs that are subject to an AMR were properly identified, the staff performed an additional review. The staff sampled the contents of the USAR based on the listing of systems and structures in LRA Tables 2.2-1 and 2.2-2 to determine whether there were systems or structures that may have intended functions as defined by 10 CFR 54.4, but were not included within the scope of license renewal.

LRA Table 3.1-1 was also consulted, since it identifies various SSCs for exclusion from the license renewal process, sometimes with explanatory annotations. These were reviewed and considered in context with the UFSAR and other sources in order to determine whether the applicant's classifications were justified and in accordance with 10 CFR 54.4(a).

The staff has reviewed and concurred with the applicant's identification of systems and structures that are to be included in the scope of license renewal. In addition to these systems and structures, the staff believes that the pressurizer spray head and the steam generator feedrings should be included in the scope of license renewal (Open Items SM-1 and SM-2, respectively). Since these components are completely enclosed by safety related, pressure boundary components, it is important to show that failures of these components could not impede certain safety related functions of the components in which they are contained (10 CFR 54.4(a)(2)).

2.2.3 Evaluation Findings

On the basis of this review, the staff concludes that *pending satisfactory resolution of Open Items SM-1 and SM-2*, there is reasonable assurance that the applicant has adequately identified the systems and structures within the scope of license renewal in accordance with the requirements of 10 CFR 54.4.

2.2.4 References

2.3 Scoping and Screening Results: Mechanical Systems

This section addresses the mechanical systems' scoping and screening results for license renewal. The mechanical systems consist of the following (the SER sections are also provided):

a) <u>Reactor Systems</u>

Reactor Coolant System Piping (2.3.1.1)
Reactor Coolant Pumps (2.3.1.2)
Pressurizer (2.3.1.3)
Reactor Pressure Vessel (2.3.1.4)
Reactor Vessel Internals (2.3.1.5)
Steam Generators (2.3.1.6)
Reactor Vessel Level Instrumentation (2.3.1.7)

2. Engineered Safety Feature Systems

Residual Heat Removal System (2.3.2.1)
Safety Injection System (2.3.2.2)

2) Auxiliary Systems

Chemical and Volume Control System (2.3.3.4)

10 CFR 54.21(a)(1) requires an applicant to identify and list structures and components subject to an AMR. These are passive, long-lived structures and components that are within the scope of license renewal. To verify that the applicant has properly implemented its methodology, the staff focuses its review on the implementation results. Such a focus allows the staff to confirm that there is no omission of mechanical system components that are subject to an AMR. If the review identifies no omission, the staff has the basis to find that there is reasonable assurance that the applicant has identified the mechanical system components that are subject to an AMR.

2.3.1 Reactor Systems

2.3.1.1 Reactor Coolant System Piping

2.3.1.1.1 Summary of Technical Information in the Application

The applicant describes the reactor coolant system piping in LRA Section 2.3.1.1 and provides a list of components subject to an AMR in LRA Table 2.3-1.

The applicant's license renewal application contains the following description of the Reactor Coolant System:

The Reactor Coolant System (RCS) consists of three similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator, a pump, loop piping, and instrumentation. The pressurizer surge line is connected to one of the loops. Auxiliary system piping connections into the reactor coolant piping are provided as necessary. The principal heat removal systems which are interconnected with the RCS are the Steam and Power Conversion, Safety Injection, and Residual Heat Removal Systems. The RCS is dependent upon the steam generators, and the steam, feedwater, and condensate systems for stored and residual heat removal from normal operating conditions to a reactor coolant temperature of approximately 35O °F.

The RCS transfers the heat generated in the core to the steam generators where steam is generated to drive the turbine generator. Borated demineralized light water is circulated at the flow rate and temperature consistent with reactor core thermal hydraulic performance requirements. The water also acts as a neutron moderator and reflector, and as a solvent for the neutron absorber used in chemical shim control. The RCS provides a boundary which contains the coolant under operating temperature and pressure conditions. It serves to confine radioactive material and limits to acceptable values its uncontrolled release to the secondary system and to other parts of the plant under conditions of either normal or abnormal reactor operation. During transient operation the systems heat capacity attenuates thermal transients

generated by the core or extracted by the steam generators. The RCS accommodates coolant volume changes within the protection system criteria.

By appropriate selection of the inertia of the reactor coolant pump (which affects pump coastdown), the thermal hydraulic effects which result from a loss of flow situation are reduced to a safe level. The layout of the system ensures natural circulation capability following a loss of flow to permit plant cooldown without overheating the core. Part of the system's piping is used by the Emergency Core Cooling System to deliver cooling water to the core during a loss-of-coolant accident.

Reactor Coolant System piping consists of piping (including fittings, branch connections, thermal sleeves, tubing, and thermowells), pressure retaining parts of valves, and bolted closures and connections. Reactor Coolant System piping is presented in two parts: (1) Class 1 piping, and (2) Non-Class 1 piping. The design code for the Reactor Coolant System piping is ASA B31.1-1955. The majority of Reactor Coolant System piping was designed to ASA B31.1; however, some small-bore piping was designed to ASME Boiler and Pressure Vessel Code, Section III.

Class 1 piping includes the Reactor Coolant System main loop piping; pressurizer surge, spray, and safety and relief valve inlet lines; vents, drains, and instrument lines. Portions of ancillary systems attached to the Reactor Coolant System are also Class 1. Ancillary systems attached to the Reactor Coolant System include the Safety Injection System, Residual Heat Removal System, Chemical and Volume Control System, and Primary Sampling System.

Several non-Class 1 piping components in the Reactor Coolant System are within the scope of license renewal for RNP:

1. The Pressurizer Relief Tank.

2. Pressurizer relief and safety valve discharge lines to the Pressurizer Relief Tank.

3. Auxiliary lines supporting RCS and PRT functions including containment isolation valves in those lines.

4. Reactor Vessel Level Instrumentation lines downstream of Class 1 boundary Bellows.

The Pressurizer Relief Tank (PRT), located inside containment, normally contains water, at or near ambient containment conditions, in a predominantly nitrogen atmosphere. Steam is discharged from relief and safety valves of the Reactor Coolant System into the Pressurizer Relief Tank where it is condensed and cooled by mixing with the water. The Pressurizer Relief Tank also collects leakage and liquid from various system pressure relief valves located inside the containment. The Pressurizer Relief Tank was designed to the ASME Boiler and Pressure Vessel Code, Section III, Class C. To reduce the likelihood of PRT overpressurization following a discharge, the PRT is equipped with a spray to add cooling water and a drain to the Waste Disposal System to remove excess heated water. The PRT is also equipped with two rupture discs that relieve pressure to the containment vessel at approximately 100 psig. The rupture discs are designed to pass 900,000 lb/hr saturated steam.

The PRT size is 1300 ft3 with a design temperature and pressure of 340°F and 100 psig respectively. The PRT is piped to the pressurizer safety and power operated relief valves by a 12 in. line. The PRT is normally filled to about 70 percent with primary water and also has approximately 3 psig nitrogen atmosphere in it. A nitrogen regulator outside containment maintains this pressure in the tank along with the ability to vent the PRT to the vent header. Primary water may be added to the tank by use of the primary water pumps and valves. Water may be pumped from the tank by utilizing the "B" reactor coolant drain tank pump and valves or gravity drained to the containment sump.

2.3.1.1.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.1, USAR Sections 5.1 and 5.4.3, and Dwg No. 5739-1971-LR (2 sheets) - Reactor Coolant System Flow Diagram, to determine whether there is reasonable assurance that the reactor coolant system piping within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review, the staff selected system functions described in the USAR that were required by 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

The applicant has included the Pressurizer Relief Tank in the pressure-retaining boundary even though this pressure-retaining boundary will be maintained only until the tank's rupture disks give way, as designed, at about 100 psid. This is acceptable to the staff, since the Pressurizer Relief Tank could play a limited role in supporting some of the functions described in 10 CFR 54.4(a)(1), particularly in situations wherein the rupture disks remain intact.

2.3.1.1.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that there is reasonable assurance that the applicant has adequately identified the reactor coolant system piping that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified reactor coolant system piping that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.3.1.2 <u>Reactor Coolant Pumps</u>

2.3.1.2.1 Summary of Technical Information in the Application

The applicant describes the reactor coolant pumps in LRA Section 2.3.1.2.

The applicant's license renewal application contains the following description of the reactor coolant pumps:

The reactor coolant pumps provide the motive force for circulating the reactor coolant through the reactor core, piping, and steam generators. Each reactor coolant loop contains a vertical single stage centrifugal pump which employs a controlled leakage seal assembly. Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the impeller, discharged through passages in the diffuser and out through a discharge nozzle in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping.

All parts of the pumps in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials. The RNP Reactor Coolant Pump casings were designed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class A.

Component cooling water is supplied to the motor bearing cooler and the thermal barrier cooling coil. The squirrel cage induction motor driving the pump is air cooled and has oil lubricated thrust and radial bearings. A water lubricated bearing provides radial support for the pump shaft. A flywheel and an anti-reverse rotation device are located at the top of the RCP motor. The flywheel provides additional inertia to increase the RCP coastdown time, thereby reducing the consequences of loss-of-coolant accident. The anti-reverse rotation device prevents backflow, which may occur during LOCA, from turning the RCP in the reverse direction.

The portion of the reactor coolant pump rotating element above the pump coupling, including the electric motor and the flywheel, is not subject to aging management review in accordance with 10 CFR 54.21(a)(1)(i). Reactor coolant pump seals are not subject to an aging management review for the following reasons:

1. Seal leakoff is closely monitored in the control room, and high leakoff flow rate is alarmed as an abnormal condition requiring corrective action,

2. The reactor coolant pump seal package and its constituent parts are periodically overhauled on a schedule established by the Preventive Maintenance Program. The seals are inspected and parts are replaced, as required.

Plant operating experience with pump seal performance has demonstrated the effectiveness of these activities.

Each RCP is supported on a three-legged structural system consisting of three connected columns fabricated of carbon steel members, structural sections and pipe. Provisions for limited movement of the structure in any horizontal direction to accommodate piping expansion are accomplished with a sliding "Lubrite" base plate arrangement and a system of tie rods and anchor bolts which restrain the structure from movement beyond the calculated limits. Sliding slot at the top of the support structures permits radial thermal growth of the pumps during heatup.

2.3.1.2.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.2 and USAR Section 5.4.1 to determine whether there is reasonable assurance that the reactor coolant pumps within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review the staff selected system functions described in the USAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

2.3.1.2.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that there is reasonable assurance that the applicant has adequately identified the reactor coolant pumps that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the reactor coolant pumps that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.3.1.3 <u>Pressurizer</u>

2.3.1.3.1 Summary of Technical Information in the Application

The applicant describes the pressurizer in LRA Section 2.3.1.3 and provides a list of components subject to an AMR in LRA Table 2.3-1.

The applicant’s license renewal application contains the following description of the pressurizer:

The pressurizer is a vertical cylindrical vessel containing electric heaters in its lower head and a water spray nozzle in its upper head. Sources of heat to the Reactor Coolant System are interconnected by piping to the pressurizer with no intervening isolation valves; the pressurizer lower head is connected to the Reactor Coolant System by the surge line. Pressure relief protection for the Reactor Coolant System is provided on the pressurizer. Overpressure protection consists of three code safety valves and two power operated relief valves. Piping attached to the pressurizer is Class 1 up to and including the safety and relief valves.

The Pressurizer was designed and fabricated in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class A. The pressurizer is constructed of carbon steel with internal surfaces clad with austenitic stainless steel. The heaters are sheathed in austenitic stainless steel. The pressurizer vessel surge nozzle is protected from thermal shock by a thermal sleeve. A thermal sleeve also protects the pressurizer spray nozzle

The pressurizer maintains the required reactor coolant pressure during steady-state operation, limits the pressure changes caused by coolant thermal expansion and contraction during normal load transients, and prevents the pressure in the Reactor Coolant System from exceeding the design pressure.

The pressurizer contains replaceable direct immersion heaters, multiple safety and relief valves, a spray nozzle and interconnecting piping, valves and instrumentation. The electric heaters located in the lower section of the vessel maintain the pressure of the Reactor Coolant System by keeping the water and steam in the pressurizer at saturation temperature corresponding to

the system pressure. Three pressurizer heater banks (1 control and 2 backup) with a total design capacity of 1300 kW are installed. A minimum total capacity of 800 kW is required for normal operating conditions. A minimum of 125 kW of heater capacity is capable of being powered from emergency power supplies. This capacity is sufficient to maintain the Reactor Coolant System near normal operating pressure and to aid natural circulation. This is automatically tripped off from the emergency bus in the event of a safety injection signal to prevent overloading of the diesel generators.

The pressurizer is designed to accommodate positive and negative surges caused by load transients. The surge line which is attached to the bottom of the pressurizer connects it to the hot leg of a reactor coolant loop. During a positive surge, caused by a decrease in plant load, the spray system, which is fed from the cold leg of a coolant loop, condenses steam in the pressurizer to prevent the pressurizer pressure from reaching the set point of the power operated relief valves. Power operated spray valves on the pressurizer limit the pressure during load transients. In addition, the spray valves can be operated manually by a switch in the control room. A small continuous spray flow is provided to assure that the pressurizer liquid is homogeneous with the coolant and to prevent excess cooling of the spray and surge line piping.

2.3.1.3.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.3 and USAR Section 15.6.3.2.1 to determine whether there is reasonable assurance that the pressurizer within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review the staff selected system functions described in the USAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

The pressurizer, a safety related, in-scope component contains a spray head, a non-safety related component, which the applicant proposes to exclude from the license renewal scope.

The spray head distributes normal and auxiliary pressurizer spray water into the pressurizer steam bubble, which tends to depressurize the pressurizer, and hence the Reactor Coolant System. Since the normal and auxiliary pressurizer sprays are not safety systems, they cannot be relied upon to function during any of the Chapter 15 accident analyses, unless, in some postulated analysis cases, pressurizer spray could have an aggravating effect upon the transient results (e.g., by delaying a high pressurizer pressure reactor trip).

However, Section 15.6.3.2.1 of the UFSAR mentions the means by which the Reactor Coolant System might be depressurized during a steam generator tube rupture (SGTR) event. The UFSAR lists, *"in order of preference: (1) normal pressurizer spray; (2) pressurizer power operated relief valves (PORVs); (3) auxiliary pressurizer spray, and; (4) balancing charging/letdown or using unaffected steam generators for cooldown/depressurization."* Normal and auxiliary pressurizer sprays are the first and third preferred means of reducing the primary side coolant pressure, and ending the primary to secondary side tube break flow. Although the spray flow rates are not determined according to any performance requirements set by the steam generator tube rupture event, the normal and auxiliary sprays constitute two of the four

listed depressurization methods. If, for some reason, the spray head fails in such a way as to block all spray flow, then normal and auxiliary sprays would become unavailable for cooldown and depressurization following a SGTR event. Since there is always some spray flow into the pressurizer, during normal operation, it is expected that such an unlikely failure would be promptly detected and rectified. Also, the applicant states that they have a procedure to deal with an SGTR, assuming the coincident loss of normal and auxiliary sprays.

Failure of the spray head by clogging (i.e., reducing the spray flow) and by dislodging the head from the end of the supply line (i.e., substituting a stream flow for the spray flow) are scenarios that are not easily evaluated, in terms of quantified depressurization effects; but these depressurization rates would be bounded by the hypothetical termination of all pressurizer spray flow. A reduction in the spray-induced depressurization rate could be seen when the normal and auxiliary pressurizer sprays are used to cooldown and depressurize following an SGTR event; but since they're not relied upon to mitigate the event, and they're not classified as safety related systems or components, they're not required to be included in the license renewal scope, under the terms of 10 CFR 54.4(a)(1).

The pressurizer spray head is a non-safety related component that is wholly enclosed by the pressurizer, a safety related component. This means the pressurizer spray head would be subject to the requirements of 10 CFR 54.4(a)(2), which states, "Plant systems, structures, and components within the scope of this part are .... All nonsafety-related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs (a)(1) (i), (ii), or (iii) of this section." Paragraphs (a)(1) (i), (ii), and (iii) of this section address the integrity of the reactor coolant pressure boundary, the capability to shut down the reactor and maintain it in a safe shutdown condition, and the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures, respectively.

If the pressurizer spray head were to degrade, and crack, and shed one or more pieces of the head, then these pieces could be considered to be loose parts inside the pressurizer. One or more of these pieces could fall away from the head either during normal operation, or during a pressurization or depressurization transient. During a pressurization transient, such as a loss or normal feedwater event, or a load rejection, the PORVs or even the code safety valves might open. A loose part, inside the pressurizer, might be drawn into the throat of a PORV or code safety valve, and impede the ability of the pressurizer and its pressure relieving valves to protect the integrity of the reactor coolant pressure boundary. Depending upon the position of the loose part, inside the valve throat, the loose part might prevent the valve from reseating properly, and thereby transform a pressurization event into a depressurization event.

During a depressurization transient, such as the spurious opening of steam dump valve, the loose part might be drawn into the pressurizer surge line, if it's not blocked by the electric immersion heaters. A piece of the pressurizer spray head would be small, compared to the inside diameter of the surge line, and not likely to block any of the pressurizer outsurge; but it might be transported into the steam generator inlet plenum, where it could hit the primary side of the tubesheet or the divider plate. If the piece is small enough, then there is also the possibility of its entry into a steam generator tube, where it can block the tube flow or damage the tube wall.

Although the breaking off of pieces from the spray head are not likely to damage the pressurizer itself, these pieces have the potential to impair certain safety related functions of the pressurizer and other safety related components during pressurization and depressurization transients. Therefore, the staff believes that the pressurizer spray head ought to be included in the license renewal scope, as per the requirements of 10 CFR 54.4(a)(2). Since the applicant has not agreed to the inclusion of the pressurizer spray head in the license renewal scope, this issue is considered to be an Open Item (designated as SM-1).

2.3.1.3.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that ***pending satisfactory resolution of Open Item SM-1***, there is reasonable assurance that the applicant has adequately identified the pressurizer SSCs that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the pressurizer SSCs that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.3.1.4 Reactor Pressure Vessel

2.3.1.4.1 Summary of Technical Information in the Application

The applicant describes the reactor pressure vessel in LRA Section 2.3.1.4 and provides a list of components subject to an AMR in LRA Table 2.3-1.

The applicant's license renewal application contains the following description of the reactor pressure vessel:

The reactor vessel consists of the cylindrical vessel shell, lower vessel head, closure head, nozzles, interior attachments, and associated pressure retaining bolting. The vessel is fabricated of a low-carbon alloy steel with austenitic stainless steel cladding on all surfaces exposed to the reactor coolant fluid. Coolant flow enters the reactor vessel through three inlet nozzles in a plane just below the vessel flange and above the core. The coolant flows downward through the annular space between the vessel wall and the core barrel into a plenum at the bottom of the vessel where it reverses direction, passes up through the core into the upper plenum and then flows out of the vessel though three exit nozzles located on the same plane as the inlet nozzles. The Reactor Pressure Vessel was designed according to the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Class A.

2.3.1.4.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.4 and USAR Section 5.3 to determine whether there is reasonable assurance that the reactor pressure vessel within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review the staff selected system functions described in the USAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

The staff agrees with the applicant's identification of the pressure vessel and its associated pressure boundary components as items that should be part of the license renewal scope.

2.3.1.4.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that there is reasonable assurance that the applicant has adequately identified the reactor pressure vessel that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the reactor pressure vessel that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.3.1.5 Reactor Vessel Internals

2.3.1.5.1 Summary of Technical Information in the Application

The applicant describes the reactor vessel internals in LRA Section 2.3.1.5 and provides a list of components subject to an AMR in LRA Table 2.3-1.

The applicant's license renewal application contains the following description of the reactor vessel internals:

The reactor vessel internals are designed to support, align, and guide the core components, and to support and guide incore instrumentation. The reactor vessel internals consists of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and a lower internals assembly that can be removed, if desired, following a complete core unload.

The lower internals assembly is supported in the vessel by resting on a ledge in the vessel head-mating surface and closely guided at the bottom by radial support/clevis assemblies. The upper internals assembly is clamped at this same ledge by the reactor vessel head. The bottom of the upper internals assembly is closely guided by the core barrel alignment pins of the lower internals assembly.

The lower internals are comprised of the core barrel, thermal shield, core baffle assembly, lower core plate, intermediate diffuser plate, bottom support plate, and supporting structures. The upper internals package (upper core support structure) is a rigid member composed of the top support plate and deep beam sections, support columns, control rod guide tube assemblies, and the upper core plate. Upon upper internals assembly installation, the last three parts are physically located inside the core barrel.

The Reactor Pressure Vessel Internals for RNP were designed prior to the creation of ASME Boiler and Pressure Vessel Code, Section III, Subsection NG. The internals were designed using internal Westinghouse design criteria that effectively evolved to become the original NG criteria. The reactor vessel internals were designed using the allowable stress levels of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, as a guide. Most of the major internal structures are fabricated from A240 Type 304 stainless steel. However, the lower support forging is A182 Type 304.

Support column assemblies are stainless steel, A213, A276, or A182 Type 304. Control rod guide tube assemblies are A240, A249, and A479 Type 304. CASS (A351 Gr. CF8) components are the upper support tube base, lower support plate columns, and bottom mounted instrumentation column cruciforms. The hold down spring is Type 304 stainless steel. The secondary core support assembly is fabricated from A240 andA479 Type 304 stainless steel; the energy absorber is made from A182 Type 304 orA336 Type F8. Clevis inserts are Alloy 600 with Inconel X-750 insert bolts. Other bolts and alignment pins are of Type 316 or Type 347 stainless steel, except the head and vessel alignment pins and radial support keys which are Type 304.

The in-core instrumentation includes in-core flux guide thimbles to permit the insertion of movable detectors for measurement of the neutron flux distribution within the reactor core. Movable miniature neutron flux detectors are available to scan the active length of selected fuel assemblies to provide remote reading of the relative three-dimensional flux distribution. The thimbles are inserted into the reactor core through guide tubes, or conduits, extending from the bottom of the reactor vessel through the concrete shield area and then up to a thimble seal table. Since the movable detector thimbles are closed at the leading (reactor) end, they are dry inside. The thimbles thus serve as a pressure barrier between the reactor coolant pressure and the atmosphere. Mechanical seals between the retractable thimbles and the conduits are provided at the seal table.

2.3.1.5.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.5 and USAR Sections 3.9.5 and 7.7.1.5 to determine whether there is reasonable assurance that the reactor vessel internals within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review the staff selected system functions described in the USAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

2.3.1.5.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that there is reasonable assurance that the applicant has

adequately identified the reactor vessel internals that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the reactor vessel internals that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.3.1.6 Steam Generator

2.3.1.6.1 Summary of Technical Information in the Application

The applicant describes the steam generator in LRA Section 2.3.1.6 and provides a list of components subject to an AMR in LRA Table 2.3-1.

The applicant’s license renewal application contains the following description of the steam generator:

The Steam Generators remove heat from the Reactor Coolant System by converting Feedwater into steam. The Steam Generators provide sufficient capacity to remove heat during normal operations and following postulated accidents and transients. An integral flow restrictor limits the flow rate of steam from a Steam Generator following a postulated steam line break accident. Steam Generator level instrumentation is provided to assure the heat removal capability is maintained following an accident.

There are three steam generators installed, one in each of the three RNP reactor coolant loops. Each steam generator is a vertical shell-and-tube heat exchanger that transfers heat from a single-phase fluid at high temperature and pressure (the reactor coolant) in the tube side, to a two-phase (steam-water) mixture at lower temperature and pressure in the shell side. The original Steam Generators (primary side) were designed to the 1965 Edition of the ASME Boiler and Pressure Vessel Code, Class A, through Summer 1965 Addenda. The shell side was designed according to the ASME Boiler and Pressure Vessel Code, Section III, Class C. The replacement Steam Generator Lower Assemblies were designed in accordance with the 1980 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Class A, through Winter 1980 Addenda.

Reactor coolant enters and exits the tube side of each steam generator through nozzles located in the lower hemispherical head. The Reactor Coolant System fluid flows through inverted U-tubes connected to the tube sheet. The lower head is divided into inlet and outlet chambers by a vertical partition plate extending from the lower head to the tube sheet. The steam-water mixture is generated on the secondary, or shell side, and flows upward through moisture separators and dryers to the outlet nozzle at the top of the vessel providing essentially dry, saturated steam. Man ways and inspection ports are provided to permit access to both sides of the lower head and to the U-tubes and moisture separating equipment on the shell side of the steam generators.

The steam generator support system includes hydraulic snubbers. The snubbers are considered to be structural components; however, portions of the hydraulic equipment for each steam generator (manifold, hydraulic control unit, flex hoses, piping, reservoir) are subject to an aging management review to assure their pressure boundary integrity is maintained.

Lower assemblies of the steam generators, including the lower shell, tubes, and tube sheet, were replaced in 1984.

2.3.1.6.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.6 and USAR Sections 5.4.2 and 10.3 to determine whether there is reasonable assurance that steam generator within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review the staff selected system functions described in the USAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

The steam generator, a safety related, in-scope component contains a feedring, a non-safety related component, which the applicant proposes to exclude from the license renewal scope.

The feedring distributes main feedwater into the steam generator shell side, through a number of J-tubes mounted along the upper surface of the feedring. The feedring is normally filled with feedwater, up to a level that is higher than the feedring itself (i.e., to a level inside the J-tubes). This arrangement prevents the formation of steam inside the feedring, which minimizes the possibility of water hammer in the feedwater system. The same feedring distributes auxiliary feedwater during startup and shutdown operations, and during certain accidents and transients.

Since the feedring is not classified as a safety related component, its inclusion in the license renewal scope is not required by 10 CFR 54.4(a)(1). Rather, it is subject to the requirements of 10 CFR 54.4(a)(2) and 10 CFR 54.4(a)(3). 10 CFR 54.4(a)(2) can be summed up, in the contrapositive, by stating that, if a nonsafety-related SSC cannot fail in such a way as to prevent the satisfactory accomplishment of the functions listed in 10 CFR 54.4(a)(1), then it need not be included in the license renewal scope. 10 CFR 54.4(a)(3) applies to all SSCs that are relied upon to perform functions necessary to comply with regulations pertaining to fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram (ATWS), and station blackout.

The feedring can fail to perform its distribution function (e.g., by clogging of some J-tubes) without materially affecting the primary to secondary heat transfer rate in the steam generator, if all the main or auxiliary feedwater flow continues to be delivered. Full flow, if not uniformly distributed, would still be adequate in the context of accident analyses, to demonstrate compliance with the applicable acceptance criteria. Therefore, clogging, or other problems that prevent the uniform distribution of main or auxiliary feedwater flowing through the feedring, would not be expected to affect any safety-related functions performed by the steam generator or associated components.

However, if failure of the feedring in other ways could possibly affect the functions listed in 10 CFR 54.4(a)(1), such as the capability to render the plant in a safe shutdown condition or to deal with accidents that could result in potential offsite exposures, then the feedring would have to be part of the license renewal scope, as per 10 CFR 54.4(a)(2). For example, if the feedring begins to degrade and crack, and a piece of the feedring or a J-tube breaks away from the

feedring and falls onto the tubesheet, it might damage the tubesheet area around the tube penetrations. A small piece might break off the feedring during a steam generator depressurization event, such as the spurious opening of a safety or dump valve. If the piece is small enough to pass through the perforated deck plate, through the steam separators, and through the flow element, then it could possibly lodge in the valve throat and damage or prevent the proper functioning of the valve. Such possibilities, though not likely, indicate that certain failures in the feedring, which could impair the safety related functions of the surrounding steam generator, could not be ruled out. Therefore, 10 CFR 54.4(a)(2) requires that the feedring be included in the scope of license renewal.

The feedring is a conduit for auxiliary feedwater, which is required, in many accident and transient scenarios, to remove decay heat. Any failure in the feedring that could result in a reduction of auxiliary feedwater delivery could prevent continued demonstration of compliance with regulations for fire protection, ATWS, or station blackout. (Clogging is not likely to be such a failure, since serious clogging would become apparent during normal operation, with a reduction in main feedwater flow.) ATWS analyses, for example, are based upon best estimate assumptions for auxiliary feedwater flow, which means all auxiliary feedwater pumps are assumed to start and deliver full rated flow. The ATWS rule, 10 CFR 50.62, requires the installation of independent circuitry to assure that a start signal is received by the auxiliary feedwater system. Thus, a common mode fault in the normal actuation circuitry could not prevent actuation of the auxiliary feedwater system, which ATWS analyses have shown is necessary in order to limit the peak pressure attained to values below the Level C Stress Limit. A feedring that somehow diminishes auxiliary feedwater flow would erode the effectiveness of the ATWS rule, which is intended to assure the availability of auxiliary feedwater. Therefore, the feedring would have to be considered to be in the license renewal scope, under the terms of 10 CFR 54.4(a)(3) unless it can be shown that there is no possible failure in the feedring that can reduce the rate of auxiliary feedwater flow delivery. The applicant has not provided such assurance.

The feedring, as it’s positioned in the downcomer of the steam generator, could provide structural support to components inside the steam generator that protect safety related components, such as the tubes.

Therefore, the staff believes that the feedring should be part of the license renewal scope. Since the applicant has not included the feedring in the license renewal scope, this is considered to be an Open Item, designated as SM-2.

2.3.1.6.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that pending satisfactory resolution of Open Item SM-2, there is reasonable assurance that the applicant has adequately identified the steam generator that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the steam generator that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.3.1.7 Reactor Vessel Level Instrumentation

2.3.1.7.1 Summary of Technical Information in the Application

The applicant describes the reactor vessel level instrumentation in LRA Section 2.3.1.7.

The applicant's license renewal application contains the following description of the reactor vessel instrumentation:

An Inadequate Core Cooling Instrumentation System is provided to detect the approach to inadequate reactor core cooling and assess the adequacy of responses taken to restore core cooling. The system consists of three subsystems: Reactor Vessel Level Instrumentation System (RVLIS), Core Exit Thermocouple System (CETS), and the Core Cooling Monitor System (CCMS). Portions of the RVLIS consist of mechanical components that are part of the Reactor Coolant System pressure boundary or part of the containment pressure boundary.

2.3.1.7.2 Staff Evaluation

The staff reviewed LRA Section 2.3.1.7 to determine whether there is reasonable assurance that the reactor vessel level instrumentation within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review the staff selected system functions described in the USAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

2.3.1.7.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that there is reasonable assurance that the applicant has adequately identified the reactor vessel level instrumentation that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the reactor vessel level instrumentation that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.3.1.8 Evaluation Findings

On the basis of this review, the staff concludes that there is reasonable assurance that the applicant has adequately identified the reactor coolant systems and components that are within the scope of license renewal, in accordance with the requirements of 10 CFR 54.4(a), and that the applicant has adequately identified the reactor coolant system components that are subject to an aging management review, in accordance with the requirements of 10 CFR 54.21(a)(1).

2.3.1.9 References

2.3.2 Engineered Safety Features Systems

2.3.2.1 Residual Heat Removal System

2.3.2.1.1 Summary of Technical Information in the Application

The applicant describes the residual heat removal system in LRA Section 2.3.2.1 and provides a list of components subject to an AMR in LRA Table 2.3-2.

The applicant's license renewal application contains the following description of the Residual Heat Removal System:

The Residual Heat Removal System delivers borated water to the Reactor Coolant System during the injection phase of a design basis accident. Following a loss-of-coolant accident, the Residual Heat Removal System cools and recirculates water that is collected in the containment recirculation sump and returns it to the Reactor Coolant, Containment Spray, and Safety Injection Systems to maintain reactor core and containment cooling functions. In addition, during normal plant operations, the Residual Heat Removal System removes residual and sensible heat from the core during plant shutdown, cooldown, and refueling operations. The RHR system is used to achieve cold shutdown conditions following a postulated fire in accordance with 10 CFR 50, Appendix R, requirements.

The Residual Heat Removal System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

3. SCs that are relied upon during postulated fires and station blackout events

2.3.2.1.2 Staff Evaluation

The staff reviewed LRA Section 2.3.2.1 and USAR Sections 5.4.4 and 6.3 to determine whether there is reasonable assurance that the residual heat removal system components within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review the staff selected system functions described in the USAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

2.3.2.1.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license

renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that there is reasonable assurance that the applicant has adequately identified the components of the residual heat removal system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the residual heat removal system that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.3.2.2 Safety Injection System

2.3.2.2.1 Summary of Technical Information in the Application

The applicant describes the safety injection system in LRA Section 2.3.2.2 and provides a list of components subject to an AMR in LRA Table 2.3-3.

The applicant's license renewal application contains the following description of the Safety Injection System:

Following a postulated design basis accident, adequate emergency core cooling is provided by the Safety Injection System, whose components operate in three modes: passive accumulator injection, active safety injection, and residual heat removal recirculation. The primary purpose of the system is to deliver cooling water to the reactor core in the event of a loss-of-coolant accident. This limits the fuel cladding temperature and thereby ensures that the core will remain intact and in place, with its heat transfer geometry preserved. The system also provides a source of borated water for reactivity control.

The Safety Injection System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs that are part of the Environmental Qualification Program

3. SCs that are relied upon during postulated fires and station blackout events

2.3.2.2.2 Staff Evaluation

The staff reviewed LRA Section 2.3.2.2 and USAR Section 6.3 to determine whether there is reasonable assurance that the safety injection system within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review the staff selected system functions described in the USAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

2.3.2.2.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that there is reasonable assurance that the applicant has adequately identified the for safety injection system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the safety injection system that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).

2.3.3.4 Chemical and Volume Control System

2.3.3.4.1 Summary of Technical Information in the Application

The applicant describes the chemical and volume control system in LRA Section 2.3.3.4 and provides a list of components subject to an AMR in LRA Table 2.3-10.

The applicant's license renewal application contains the following description of the Chemical and Volume Control System:

The Chemical and Volume Control System provides a continuous feed and bleed of reactor cooling water for the Reactor Coolant System to maintain proper water level and to adjust boron concentration. The Chemical and Volume Control System provides a means for injection of control poison in the form of boric acid solution, chemical additions for corrosion control, and reactor coolant cleanup and degasification. The system also adds makeup water to the RCS, reprocesses water letdown from the RCS and charging pump leakage, and provides seal water injection to the reactor coolant pump seals. Portions of the CVCS contain highly concentrated boric acid solution and may be used to maintain reactor Shutdown Margin in accordance with Technical Specification requirements. However, providing highly concentrated boric acid solution is not a safety related function; and the components that provide this solution are not inscope of license renewal unless they provide another function that is in scope. The license renewal evaluation boundaries for the Chemical and Volume Control System are shown on the following flow diagrams. (Flow diagrams have been submitted separately for information only.)

The Chemical and Volume Control System is in the scope of license renewal, because it contains:

1. SCs that are safety-related and are relied upon to remain functional during and following design basis events

2. SCs which are non-safety related whose failure could prevent satisfactory accomplishment of the safety related functions

3. SCs that are part of the Environmental Qualification Program

4. SCs that are relied on during postulated fires and station blackout events

2.3.3.4.2 Staff Evaluation

The staff reviewed LRA Section 2.3.3.4 and USAR Section 9.3.4 to determine whether there is reasonable assurance that the chemical and volume control system components within the scope of license renewal and subject to an AMR have been identified in accordance with 10 CFR 54.4 and 54.21(a)(1).

In the performance of the review the staff selected system functions described in the USAR that were set forth in 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the Rule. The staff also focused on components that were not identified as being subject to an AMR to determine if any components were omitted.

2.3.3.4.3 Conclusions

The staff reviewed the LRA and the accompanying scoping boundary drawings to determine whether any structures, systems, or components that should be within the scope of license renewal were not identified by the applicant. No omissions were found. In addition, the staff performed an independent assessment to determine whether any components that should be subject to an AMR were not identified by the applicant. No omissions were found. On the basis of this review, the staff concludes that there is reasonable assurance that the applicant has adequately identified the components of the chemical and volume control system that are within the scope of license renewal, as required by 10 CFR 54.4(a), and that the applicant has adequately identified the components of the chemical and volume control system that are subject to an aging management review, as required by 10 CFR 54.21(a)(1).