1. INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This document is a safety evaluation report (SER) on the application to renew the operating licenses for McGuire Nuclear Station, Units 1 and 2 (McGuire or McGuire 1 and 2), and Catawba Nuclear Station, Units 1 and 2 (Catawba or Catawba 1 and 2), filed by Duke Energy Corporation (Duke or the applicant). Throughout this SER, “McGuire” or “Catawba” refers to both units (unit 1 and unit 2). When the staff discusses information specific to a particular unit, it will refer to that unit as McGuire 1, McGuire 2, Catawba 1, or Catawba 2.

By letter dated June 13, 2001, Duke submitted its application to the United States Nuclear Regulatory Commission (NRC) for renewal of the McGuire and Catawba units’ operating licenses for up to an additional 20 years. The application was received by the NRC on June 14, 2001. The NRC staff reviewed the McGuire and Catawba 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,” and prepared this report to document its findings. The project manager for the McGuire and Catawba safety review is Rani Franovich. Ms. Franovich may be contacted by telephone at (301) 415-1868 or by electronic mail at rlf2@nrc.gov. Alternatively, written correspondence can be sent to the following address:

License Renewal and Environmental Impacts Program
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
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In its LRA, the applicant requested renewal of the operating licenses issued under Section 103 of the Atomic Energy Act of 1954, as amended, for McGuire 1 and 2 (License Nos. NPF-9 and NPF-17) and Catawba 1 and 2 (License Nos. NPF-35 and NPF-52). For McGuire 1, Duke requested a period of 20 years beyond the current license expiration date of June 12, 2021.

The current operating licenses for McGuire 2, Catawba 1, and Catawba 2 expire on March 3, 2023, December 6, 2024, and February 24, 2024, respectively. Duke had requested, by letters dated June 22, 1999, an exemption from 10 CFR 54.17(c), which prohibits an applicant for renewal from submitting its application earlier than 20 years before the expiration of it’s current operating license. By letters dated October 1, 2001, the NRC staff issued exemptions from this requirement for McGuire 2, and Catawba 1 and 2 with the safety evaluation reports enclosed. Therefore, in its license renewal application, Duke requested a period of 40 years from the date of the issuance of the renewed licenses for McGuire 2 and Catawba 1 and 2, which is less than 20 years beyond the current license expiration dates for these units.

The McGuire plant is located in northwestern Mecklenburg County, North Carolina, 17 miles north-northwest of Charlotte, North Carolina. Both McGuire units consist of Westinghouse pressurized-water reactors with nuclear steam supply systems designed to operate at core power levels up to 3411 megawatts thermal, or approximately 1129 megawatts electric. Details concerning the plant and the site are found in the updated final safety analysis report (UFSAR) for McGuire.

The Catawba plant is located in the north central portion of South Carolina, in northeastern York County, approximately 18 miles southwest of Charlotte, North Carolina. Both Catawba units consist of Westinghouse pressurized-water reactors with nuclear steam supply systems designed to operate at core power levels up to 3411 megawatts thermal, or approximately 1129 megawatts electric. Details concerning the plant and the site are found in the UFSAR for Catawba.

The license renewal process proceeds along two tracks: a technical review of safety issues and an environmental review. The requirements for these two reviews are stated in NRC regulations 10 CFR Parts 54 and 51, respectively. The safety review is based on Duke's LRA and on the applicant's answers to requests for additional information (RAIs) from the NRC staff. In meetings and docketed correspondence, Duke has also supplemented its answers to the RAIs. The public can review the LRA and all pertinent information and material, including the UFSAR, at the NRC Public Document Room, 11555 Rockville Pike, Rockville, MD 20852-2738. In addition, the McGuire and Catawba LRA and significant information and material related to the license renewal review are available on the NRC web page at www.nrc.gov.

This SER summarizes the findings of the staff's safety review of the McGuire and Catawba LRA and describes the technical details considered in evaluating the safety aspects of the proposed operation of the plants for up to an additional 20 years beyond the term of the current operating licenses. The staff reviewed the LRA in accordance with NRC regulations and the guidance presented in the NRC "Standard Review Plan (SRP) for the Review of License Renewal Applications for Nuclear Power Plants," which was issued as NUREG-1800 in July 2001.

Chapters 2 through 4 of the SER document the staff's review and evaluation of license renewal issues that have been considered during the review of the LRA. Chapter 5 is reserved for the report of the Advisory Committee on Reactor Safeguards (ACRS). Appendix A is a chronology of NRC's and the applicant's principal correspondence related to the review of the LRA. Appendix B is a bibliography of the documents used during the review. The NRC staff's principal reviewers for this project are listed in Appendix C.

In accordance with 10 CFR Part 51, the staff prepared a draft plant-specific supplement to the generic environmental impact statement (GEIS). The supplement discusses the environmental considerations related to renewing the licenses for McGuire and Catawba. The draft plant-specific supplement to the GEIS was issued separately from this report. Specifically, NUREG-1437, Supplement 8, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding McGuire Nuclear Station, Units 1 and 2," issued May 2002, is the draft environmental impact statement for McGuire. Similarly, NUREG-1437, Supplement 9, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding Catawba Nuclear Station, Units 1 and 2," issued May 2002, is the draft environmental impact statement for McGuire.

1.2 License Renewal Background

Pursuant to the Atomic Energy Act of 1954, as amended, and NRC regulations, licenses for commercial power reactors to operate are issued for up to 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, not technical limitations. However, some individual plant and equipment designs may have been engineered on the basis of an expected 40-year service life.

In 1982, the NRC anticipated interest in license renewal and held a workshop on nuclear power plant aging. That led the NRC to establish a comprehensive program plan for nuclear plant aging research (NPAR). On the basis of the results of that 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 comment on a policy statement that would address major policy, technical, and procedural issues related to life extension for nuclear power plants.

In 1991, the NRC published the 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 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 programs, particularly for the implementation of the maintenance rule, which also manages plant aging phenomena.

As a result, in 1995 the NRC amended the license renewal rule. The amended 10 CFR Part 54 established a regulatory process that is expected to be 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 rule changes were intended to ensure that important systems, structures, and components (SSCs) will continue to perform their intended functions in the period of extended operation. In addition, the integrated plant assessment (IPA) process was clarified and simplified to be consistent with the revised 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 review of environmental impacts of license renewal and to fulfill, in part, the NRC’s responsibilities under the National Environmental Policy Act of 1969 (NEPA).

1.2.1 Safety Reviews

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 provides and maintains an acceptable level of safety, with the possible exception of the detrimental effects of aging on the functionality of certain SSCs 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, the rule (in 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, anticipated transients without scram, and station blackout.

Pursuant to 10 CFR 54.21(a), the applicant must review all SSCs that are within the scope of the rule to identify 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 specified time period. As required by 10 CFR 54.21(a), the applicant must demonstrate that the effects of aging will be managed in such a way that the intended function or functions of the SCs that are 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 that may affect active equipment are more readily detectable and will be identified and corrected through routine surveillance, performance monitoring, and maintenance activities. The surveillance and maintenance programs and activities for active equipment, as well as other aspects of maintaining the plant design and licensing basis, are required to continue throughout the period of extended operation.

Pursuant to 10 CFR 54.21(d), each LRA is required to include a supplement to the final safety analysis report (FSAR). This FSAR supplement must contain a summary description of the applicant's 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 several 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 on these SSCs will be adequately managed for the period of extended operation.

In July 2001, the NRC issued Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating License"; NUREG-1800, "Standard Review Plan for the Review of License Renewal Application for Nuclear Power 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 techniques used by the NRC staff in evaluating applications for license renewal. The draft versions of these documents were issued for public comment on August 31, 2000 (64 FR 53047). The staff assessment of public comments was issued in July 2001 as NUREG-1739, "Analysis of Public Comments on the Improved License Renewal Guidance Documents." The regulatory guide endorsed an implementation guideline prepared by the Nuclear Energy Institute (NEI) as an acceptable method of implementing the license renewal rule. The NEI guideline is NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Revision 3, issued in

March 2001. The regulatory guide will be used along with the SRP to review this LRA and to assess topical reports on license renewal submitted by industry groups. As experience is gained, the NRC will improve the SRP and clarify the regulatory guidance.

1.2.2 Environmental Reviews

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, Revision 1), to document its evaluation of the possible environmental impacts associated with 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. Analyses of environmental impacts of license renewal that must be evaluated on a plant-specific basis 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 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 not considered in the GEIS. Four public meetings were held, two near McGuire on September 25, 2001, and two near Catawba on October 23, 2001, as part of the NRC's scoping process to identify environmental issues specific to the plant. The results of the environmental review and a preliminary recommendation on the license renewal action were documented in NRC draft plant-specific Supplements 8 and 9 to the GEIS, which were issued on May 6, 2002, and May 13, 2002, for McGuire and Catawba, respectively. Four additional public meetings have been conducted, two near McGuire on June 12, 2002, and two near Catawba on June 27, 2002 (during the 75-day comment period for draft plant-specific Supplements 8 and 9 to the GEIS). At the meetings, the staff described the environmental review and answered questions from members of the public to help them formulate their comments on the review. Final Supplements 8 and 9 to the GEIS are scheduled to be issued on January 17, 2003.

Draft Supplements 8 and 9 to the GEIS present the NRC's preliminary environmental analysis of the effects of renewing the McGuire and Catawba operating licenses for up to an additional 20 years. The analysis considers and weighs the environmental effects and alternatives that are available to avoid adverse environmental effects. On the basis of analyses and findings in the "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants" (NUREG-1437), the environmental reports submitted by the applicant, consultation with other Federal, State, and local agencies, its own independent review, and its consideration of public comments, the staff recommended in Supplements 8 and 9 to NUREG-1437 that the Commission determine that the adverse environmental impacts of license renewal for McGuire and Catawba are not so great that preserving the option of license renewal for energy planning decisionmaking would be unreasonable.

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 McGuire and Catawba LRA in accordance with Commission guidance and the requirements of 10 CFR 54.19, 54.21, 54.22, 54.23, and 54.25. The standards for renewing a license are contained in 10 CFR 54.29.

In 10 CFR 54.19(a), the Commission requires a license renewal applicant to submit general information. Duke submitted this general information in Chapter 1 of its application for renewal of the McGuire and Catawba operating licenses. In 10 CFR 54.19(b), the Commission requires that LRAs 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." The applicant states the following in Section 1.6 of its LRA regarding this issue:

> The current indemnity agreement for McGuire Nuclear Station (B-83) 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 Attachment to the indemnity agreement, as revised through Amendment No. 10, lists NPF-9 and NPF-17, the license numbers for McGuire Nuclear Station Units 1 and 2, respectively. Should the license numbers be changed upon issuance of the renewed licenses, Duke requests that conforming changes be made to Item 3 of the Attachment to Indemnity Agreement B-83, and any other sections of the indemnity agreement as appropriate.
>
> The current indemnity agreement for Catawba Nuclear Station (B-100) 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 Attachment to the indemnity agreement, as revised through Amendment No. 9, lists NPF-35 and NPF-52, the license numbers for Catawba Nuclear Station Units 1 and 2, respectively. Should the license numbers be changed upon issuance of the renewed licenses, Duke requests that conforming changes be made to Item 3 of the Attachment to Indemnity Agreement B-100, and any other sections of the indemnity agreement as appropriate.

The staff will use the original license number for the renewed license. 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 integrated plant assessment (IPA); (b) current licensing basis changes during NRC review of the LRA; (c) an evaluation of TLAAs; and (d) an FSAR supplement. The applicant submitted the information required by 10 CFR 54.21(a), (c), and (d) in the Technical Information volume of the LRA. By letter dated June 25, 2002, the applicant submitted Amendment 1 to the LRA, which summarizes changes to the current licensing basis that have occurred at McGuire and Catawba during the staff's review of the LRA. This submittal satisfies the requirement of 10 CFR 54.21(b) and is still under staff review.

In 10 CFR 54.22, the Commission states requirements regarding technical specifications. In Appendix D of the LRA, the applicant stated that no technical specification changes had been identified as being necessary to support issuance of the renewed operating licenses for McGuire 1 and 2 and Catawba 1 and 2.

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

The staff's evaluation of the environmental information required by 10 CFR 54.23 is documented in the draft plant-specific supplements to the GEIS (NUREG-1437, Supplements 8 and 9).

1.3.1 Westinghouse Topical Reports

In accordance with 10 CFR 54.17(e), the applicant references certain Westinghouse Owners Group topical reports in each LRA. The applicant used topical reports to generically demonstrate that applicable aging effects for reactor coolant system components will be adequately managed for the period of extended operation.

- WCAP-14535A, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," Section 4.3.1, Westinghouse Electric Corporation, November 1996.

- WCAP-10456, "The Effects of Thermal Aging on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse Nuclear Steam Supply Systems," Westinghouse Electric Corporation, November 1983.

- WCAP-10585, "Technical Basis For Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis For McGuire Units 1 and 2," June 1984, Westinghouse Electric Corporation.

- WCAP-10546, "Technical Basis For Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis For Catawba Units 1 and 2," June 1984, Westinghouse Electric Corporation.

The staff issued the safety evaluation for WCAP-14535A on September 12, 1996. In accordance with the procedures provided in NUREG-0390, "Topical Report Review Status," the staff requested that the Westinghouse Owners Group publish the accepted versions of the reports incorporating the transmittal letter and the staff's safety evaluation between the title page and the abstract. The accepted versions have an *A* (for "accepted") after the report identification number.

The safety evaluations of the topical reports are intended to be stand-alone documents. An applicant incorporating the topical reports by reference into its 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 LRA is documented in Chapter 4 of this SER.

1.4 Summary of Open Items and Confirmatory Items

<u>Open items 2.3-1 and 2.3-2.</u> The applicant failed to perform an AMR for the housings of active components (e.g., fans and dampers) that may perform critical pressure retention and/or structural integrity functions. Failure to maintain that function could prevent the associated active component from performing its function. Since these housings are within the scope of license renewal and are long-lived and passive, they are subject to an AMR in accordance with 10 CFR 54.21.

Open item 2.3-3. The AMP (the Inspection Program for Civil Engineering Structures and Components) credited by the applicant for monitoring the aging of structures that include structural sealants as sub-components does not include, within its scope, building sealants. Therefore, this AMP is not adequate to manage the aging of building sealants, which are long-lived, passive structural sub-components within the scope of license renewal.

Open item 2.3.3.12.2-1. By letter dated January 28, 2002, the staff requested, in RAI 2.3.3.12-1, that the applicant provide the basis for not listing the turbocharger turbine flexible hose in Table 3.3-15, since these components are passive, long-lived, and have pressure boundary intended functions. In its response, dated April 15, 2002, the applicant stated that the flexible hose is replaced during periodic maintenance. The applicant implied that the hose is replaced based on qualified life in accordance with 10 CFR 54.21(a)(1)(i) and is, therefore, not subject to an AMR. However, since this was not clearly stated in the RAI response, this issue is characterized as an open item.

Open item 2.3.3.13.2-1. The applicant did not provide sufficient information in its response to RAI 2.3.3.13-1 to enable the staff to evaluate the adequacy of its replacement of synthetic rubber flexible expansion joints associated with the emergency diesel generator crankcase vacuum system during periodic maintenance. The applicant should indicate if replacement of these components is based upon a qualified life or based upon condition or performance monitoring. If replacement is based upon the latter, the applicant should specify the parameters that will be monitored as indicators of the components' condition or performance.

Open item 2.3.3.14.2-1. The applicant did not provide sufficient information in its response to RAI 2.3.3.14-1 to enable the staff to evaluate the adequacy of its replacement of flexible hose connections associated with the emergency diesel generator fuel oil system during periodic maintenance. The applicant should indicate if replacement of these components is based upon a qualified life or based upon condition or performance monitoring. If replacement is based upon the latter, the applicant should specify the parameters that will be monitored as indicators of the components' condition or performance.

Open item 2.3.3.19-1. McGuire UFSAR Section 9.5.1.2.1 states that fire hydrants are connected to the yard main. Furthermore, fire hydrants are considered passive and long-lived components in accordance with 10 CFR 54.21. Since the UFSAR is referenced in the license conditions for both McGuire and Catawba, and these components are discussed therein as providing a fire suppression function (which is required by 10 CFR 50.48), it appears that these components are required to meet the requirements of 10 CFR 50.48. The UFSAR does not distinguish between those fire hydrants that are required by 10 CFR 50.48 and those that are not. McGuire is required to meet Appendix A to BTP 9.5-1 and Catawba is required to meet the position documented in CMEB 9.5-1. Accordingly, both documents state that "outside manual hose installation should be sufficient to reach any location with an effective hose stream. To accomplish this, hydrants should be installed approximately every 250 feet on the yard main system." Therefore, the applicant should furnish documentation that demonstrates that the excluded fire hydrants are not required by 10 CFR 50.48 or identify these hydrants as being within the scope of license renewal and subject to an AMR.

Open item 2.3.3.19-2. Operating license conditions for McGuire and Catawba, Supplement 2 of the McGuire and Catawba Safety Evaluation Reports (SERs) for original licensing, and Section 9.5.1.2.1 of the McGuire and Catawba UFSARs indicate that jockey pumps are provided to prevent frequent starting of the fire pumps by maintaining pressure in the yard mains in accordance with Section 6.b of BTP CMEB 9.5-1and NFPA 20. The staff is concerned that the applicant has misapplied the QA Condition 3 designation for license renewal scoping purposes and excluded jockey pumps from the scope of license renewal, although the licensing basis of the plants indicates that these jockey pumps are relied upon to perform a function required by 10 CFR 50.48.

Open item 2.3.3.19-3. Duke did not identify Catawba fire suppression equipment to lower containment carbon filters as within the scope of license renewal. Section 9.5.1.2.1 of the UFSAR states that the RF system provides a fixed water suppression system for charcoal filters. On pages 48-50 of Duke's revised response to Appendix A to BTP APCSB 9.5-1, submitted to the NRC by letter dated November 4, 1983, Duke stated that lower containment carbon filters are provided with fire suppression capability. According to NRC Inspection Report 50-369/02-05, 50-370/02-05, 50-413/02-05 and 50-414/02-05 (ML021280003), Duke Specification CNS-1465.00-00-0006 states that carbon filters are protected by built-in water spray systems. The staff does not believe that the applicant's distinction between charcoal and carbon filters is material. Therefore, the applicant should identify water suppression equipment associated with the protection of carbon (or charcoal) filters as within the scope of license renewal.

Open item 2.3.3.19-4. A license condition for McGuire and Catawba states that Duke Energy Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the UFSARs for the respective facilities. Sections 9.5.1.2.1 and 9.5.1.2.2 of the UFSARs state that manual hose stations and automatic sprinkler or deluge systems are provided for the protection of oil storage house; the oxygen and acetylene gas storage yard area; compressed flammable gas cylinder storage area; main turbine piping and bearings; unit start-up and standby oil-filled power transformers; main turbine lube oil reservoirs; hydrogen seal oil unit; and the feedwater pump turbines. The UFSARs do not differentiate between those manual hose station and automatic sprinklers that are required to comply with 10 CFR 50.48 and those that are not. Additionally, the regulations governing fire protection apply to more than the protection of structures and equipment relied upon for safe plant shutdown. Therefore, the applicant should furnish documentation that demonstrates that the fire protection features are not required by 10 CFR 50.48 or identify the components associated with these manual hose stations and automatic sprinkler or deluge systems as being within the scope of license renewal and subject to an AMR.

Open item 2.3.3.19-5. The staff agrees with the applicant that the strainers perform an intended function that meets one of the scoping criteria (specifically 10 CFR 54.4(a)(3). The staff's technical concern is that Duke uses lake water to supply their fire protection suppression systems at McGuire and Catawba. Lake water is corrosive and may contain sediment, which can potentially clog the fire pumps. In addition, the strainers keep debris from plugging the sprinkler nozzles in fire suppression systems in the event that sprinklers are actuated. This FP component should be managed in an AMP. However, the staff is concerned that the strainers were inappropriately screened out. Although the strainers may be in-line with and connected to the main fire pump, their function is passive (as is the pump casing's function). The applicant included the pump casings within the scope of license renewal; the strainers also should be within scope.

New open Item 2.3.3.19-6. 10 CFR 50.48 requires each operating nuclear station to have a fire protection plan. A license condition for McGuire and Catawba states that Duke Energy Corporation shall implement and maintain in effect all provisions of the approved fire protection program as described in the UFSAR for the respective facilities. Section 9.5.1.2.3, "Fire Protection, Category I Safety Related," of the McGuire UFSAR states that the manually operated water spray systems provide fixed spray patterns of water for Reactor Building Purge Exhaust Filters 1A, 1B, 2A and 2B. However, drawing MCFD 1599-02.01, coordinates H-3, G-3, C-5 and B-7, indicates that piping and sprinklers associated with this function are also excluded from scope. The staff is concerned that the manually operated water spray systems for these filters were inappropriately excluded from the scope of license renewal and an AMR.

Open item 2.3.3.35.2-1. The applicant did not provide sufficient information in its response to RAI 2.3.3.35-3 to enable the staff to evaluate the adequacy of its replacement of flexible hose connections associated with the standby shutdown diesel generator fuel oil sub-system during periodic maintenance. The applicant should indicate if replacement of these components is based upon a qualified life or based upon condition or performance monitoring. If replacement is based upon the latter, the applicant should specify the parameters that will be monitored as indicators of the components' condition or performance.

Open item 2.5-1. By letter dated June 26, 2002, the applicant provided AMR results for the passive, long-lived structures and components associated with the offsite power path. Pending completion of the staff's review of this information, this item is characterized as open.

New open item 3.0.3.2.3-1. The applicant provided in Appendix A-1 (McGuire) and A-2 (Catawba) new FSAR sections describing the chemistry control program. The information provided for the FSAR is consistent with the program described in Appendix B; however, the applicant should include a discussion in the FSAR Supplement regarding the specific technical specifications and the EPRI guidelines that are mentioned in Appendix B for the chemistry control program.

New open item 3.0.3.9.1.2(a-g). The applicant's acceptance criteria for heat exchanger preventive maintenance are not adequate to provide the staff with reasonable assurance that loss of material of the heat exchanger components will be adequately managed or monitored such that the intended functions of the heat exchangers will be maintained during the extended period of operation. This open item applies to seven aging management activities (a through f).

New open item 3.0.3.10.2-1. The staff believes that volumetric examination of a sample of small-bore Class-1 piping is needed to demonstrate that the effects of aging are being adequately managed. Volumetric examination techniques provide a demonstrated capability and a proven industry record to permit detection and sizing of significant cracking and flaws in piping weld and base material. The sample of affected welds selected for inspection should be based upon piping geometry, pipe size and flow conditions, and the inspection should be performed by qualified personnel using approved station procedures.

New open item 3.0.3.10.2-2. In October 2000, a through-wall crack was identified in the reactor vessel hot leg piping at V. C. Summer. Specifically, the crack was located in the first weld between the reactor vessel nozzle and the "A" loop hot leg piping, approximately 3 feet from the reactor vessel and 7 degrees clockwise from the top dead center of the weld (as viewed from the centerline of the reactor vessel). The weld was fabricated from Alloy 82/182 material. The failure mode was determined to be primary water stress corrosion cracking and the root cause of the cracking was attributed to the presence of high residual stresses resulting from extensive repairs of the subject weld. The staff requests the applicant to identify the locations in the McGuire and Catawba RCS piping that contain welds fabricated from Alloy 82/182 material. Additionally, the staff requests the applicant to describe the actions it plans to take to address this operating experience as it applies to McGuire and Catawba.

New open item 3.0.3.11.3-1. The UFSAR supplements do not include reference to several of the important industry codes and standards discussed in the applicant's March 11, 2002, response to the staff's RAIs on the Inspection Program for Civil Engineering Structures and Components. The FSAR Supplement should be updated to reflect these codes and standards.

New open item 3.0.3.13.2-1. In the case of the buried piping, the staff finds the applicant's Preventive Maintenance Activities - Condenser Circulating Water System Internal Coating Inspection program ineffective at revealing degradation of the external pipe surface before the component pressure boundary is breached and leakage occurs. The staff believes that the applicant should propose an activity to verify that the external surfaces of buried components are not degrading based upon some sampling assessment of most vulnerable locations.

New open item 3.0.3.15.2-1. In its description of the Service Water Piping Corrosion program, Monitoring and Trending element, the applicant stated that localized corrosion due to pitting and MIC will reveal itself through pinhole leaks in the piping components, that they are not a structural integrity concern, and that they cannot individually lead to loss of the component intended function, since sufficient flow at prescribed pressures can still be provided by the system. The applicant also state that these localized concerns will lead to structural integrity concerns only when a significant number of pinholes are present and that a trend of indications of through-wall leaks will trigger corrective actions. However, the staff believes that localized corrosion can result in the loss of pressure boundary intended function under a design basis event before the corrosion reveals itself as pinhole leaks. Therefore, the applicant should justify how its program will manage the effects of localized corrosion from pitting and MIC to ensure that the intended pressure boundary function can be maintained under all design basis events consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(3).

New open item 3.0.3.18.3-1. The FSAR supplements do not include reference to some important industry standards and the NRC guidelines used for the Underwater Inspection of Nuclear Service Water Structures program. The UFSAR Supplement should be updated to reflect these standards and guidelines.

New open item 3.1.2.2.2-1. Under the Monitoring and Trending element of the Pressurizer Spray Head Examination, the applicant stated that a visual examination (VT-3) would be performed, and that no actions are taken as part of this program to trend inspection or test results. However, the staff's position is that VT-3 examinations may not be capable of detecting cracks that may occur in the pressurizer spray head. The staff therefore requests that the applicant amend the Pressurizer Spray Head Examination to state that VT-1 examination methods, which are capable of detecting and resolving cracks in the pressurizer spray heads, will be used for the one-time inspection. The scope of this open items includes the potential need to revise the acceptance criteria for this program and the FSAR Supplement summary description.

New open item 3.1.3.2.2-1. The staff reviewed the surveillance capsule schedules in Tables B.3.26-1 and B.3.26-2 of the LRA. For McGuire 1, capsule “W” is a stand-by capsule and would be withdrawn at a fluence that is significantly above the equivalent of 60 years. The applicant needs to remove this capsule and place it in storage to prevent further exposure and preserve its ability to provide meaningful metallurgical data. For Catawba 2, capsule “U” is a stand-by capsule. It appears to the staff that this capsule should be inserted in the reactor vessel and begin to accumulate fluences in an operating environment for data collection purposes. The staff believes that the applicant should place all pulled capsules in storage so that they may be saved for future use. In addition, after the applicant has pulled all the capsules, it should use alternative dosimetry to monitor neutron fluence during the period of extended operation. The applicant needs to discuss its plans for this capsule with the staff.

New open item 3.1.3.2.2-2. The staff and nuclear power industry are pursuing resolution of the reactor vessel penetration nozzle cracking issue associated with the Davis Besse boric acid corrosion and reactor vessel head wastage issue identified in October 2000. The staff is evaluating potential changes to the requirements governing inspections of Alloy 600 vessel head penetration (VHP) nozzles and PWR upper RV heads (specifically with respect to non-destructive examinations and the ability to detect cracking in the VHP nozzles prior to loss of material in the upper RV heads). This is an emerging issue that has not yet been resolved and is beyond the scope of this license renewal review, pursuant to 10 CFR 54.30(b). However, since this issue might not be resolved prior to issuance of the renewed operating licenses for the McGuire and Catawba units, the staff requests the applicant to commit to implementing any actions, as part of the VHP Nozzle Program, that are agreed upon between the NRC, NEI, MRP, and the nuclear power industry to monitor for, detect, evaluate, and correct cracking the VHP nozzles of U.S. PWRs, specifically as the actions relate to ensuring the integrity of VHP nozzles in the McGuire and Catawba upper RV heads during the extended period of operation. This commitment will ensure that the applicant’s VHP Nozzle Program (as described in the McGuire and Catawba UFSARs) will be capable of monitoring for, detecting, evaluating, and correcting cracking in the McGuire and Catawba VHP nozzles and associated upper RV heads before unacceptable degradation of the VHP nozzles or associated upper RV heads occurs. Any updates to the VHP Nozzle Program that result from resolution of this issue should be reflected in the McGuire and Catawba UFSARs.

New open item 3.1.4-1(a). Since the fabricator for the McGuire 1 and Catawba 2 RVs is not the same as the design fabricators for McGuire 2 and Catawba 1 RVs or for the Oconee RVs, some uncertainty exists whether the inspections of welded RV internals at Oconee 1 and McGuire 1 will be truly representative of the condition of welded RV internals at McGuire 2 and the Catawba units. The staff’s position is that the applicant should schedule inspection of remaining RV internal plates, forgings, welds and bolts (i.e., core barrel bolts and thermal shield bolts) at all of the McGuire and Catawba reactor units.

New open item 3.1.4-1(b). The critical crack size acceptance criterion for RV internal forgings, plates, and welds, and RV internals made from CASS have not yet been established. Nor have any acceptance criteria been proposed for the inspections that might be proposed to monitor the RV internals for void swelling. The applicant will need to submit the critical crack size acceptance criteria for the RV internal forgings, plates, and welds, and RV internals made from CASS once the evaluations for these components have been completed and the critical crack sizes for these components have been established. Once the applicant has finalized its evaluation of void swelling of the RV internals, the applicant will also need to submit the acceptance criteria for dimensional changes that might result in the RV internal components as a result of void swelling.

New open item 3.1.5-1. The staff requests the applicant to include a reference to NEI 97-06 in a summary description of the Steam Generator Surveillance Program or in Tables 18-1 of the McGuire and Catawba FSAR Supplements.

New open item 3.3.6.2.1-1. In its response to RAI 2.3.3.6-6, the applicant provided the AMR results for condenser circulating water system expansion joints at Catawba. The material for these expansion joints was specified as synthetic rubber coated with chlorobutyl rubber; the environment was specified as the yard. The applicant did not identify any aging effects; nor did the applicant specify any AMP for these components. However, the staff concluded that exposure of these expansion joints to ultraviolet (UV) rays could cause degradation over time, Because the applicant's description of the yard environment in the LRA did not address sun exposure, the staff was unable to verify that there are no applicable aging effects for these components. The applicant needs to submit a more detailed description of the yard environment for the condenser circulating water system expansion joints to address UV exposure.

New open item 3.3.17.2.1-1. In its response to RAI 2.3.3.17-2, the applicant provided the AMR results for a carbon steel emergency diesel generator starting air distributor filter in a sheltered environment. The applicant indicated that no aging effects were identified for this component. However, the staff noted that this conclusion was not consistent with the applicant's treatment of other carbon steel components in a sheltered (moist air) environment that are listed in Table 3.3-23, "Aging Management Review Results - Diesel Generator Starting Air System (McGuire Nuclear Station)." The applicant needs to explain why the carbon steel emergency diesel generator starting air distributor filter in a sheltered environment is not subject to loss of material or identify this aging effect and an AMP to manage or monitor the associated loss of material.

Open item 3.3.35.2-1. The staff requested additional information pertaining to Table 3.3-44, "Aging Management Review Results - Standby Shutdown Diesel Generator." This table indicates that the cooling water and jacket water engine radiator heat exchanger has a heat transfer function that is managed by the Chemistry Control Program. Heat transfer monitoring is not identified as a capability of the Chemistry Control Program, as defined in Appendix B, Section B.3.6. The applicant was requested to explain how the Chemistry Control Program monitors the heat transfer function. In its response, the applicant stated that for the heat exchangers in the standby shutdown diesel generator cooling water and jacket water heating sub-system, fouling would not occur because there is constant flow through the heat exchangers and because the treated water in the system is filtered to remove particles. Therefore, no aging management program is required. The staff does not agree with the applicant's conclusion that fouling will not occur in the heat exchanger because of the constant flow through the heat exchanger. The staff recognizes that sufficient flow through the heat exchanger may prevent areas of stagnation in which fouling may occur. However, the applicant has not substantiated its conclusion with any operating experience, such as maintenance and surveillance results, that reflect the success of this activity in preventing fouling. With respect to the filtering of the treated water to remove particles, the staff recognizes that particulates are removed through a filtering process. However, the applicant did not list or credit a periodic surveillance of the filter to ensure that the entrained particles do not create a high differential pressure and adversely affect flow through the heat exchanger.

New open item 3.4.1.2.2-1. The applicant proposes to mitigate general corrosion and loss of material of the auxiliary feedwater system carbon steel piping components by chemistry control. However, the staff believes that the effectiveness of the Chemistry Control program should be verified by implementing a one-time inspection of the internal surfaces of these components.

Open item 3.5-1. Contrary to the applicant's claim that aging management of concrete components via periodic inspections is only necessary for concrete SCs that are exposed to harsh environments, the staff's position is that both the operating and environmental conditions, as well as the aging of concrete nuclear components, are subject to change throughout the period of extended operation. Therefore, the applicant needs to periodically inspect these components. Although the applicant has performed an aging management review pursuant to 10 CFR 54.21(a)(3) for each structure and component that was determined to be in the scope of license renewal, the staff position (issued by letters dated November 23, 2001 [ML013300426], and April 5, 2002 [ML020980194]) is that aging management reviews should be used to differentiate between those components requiring only periodic inspections and those requiring further evaluation. Aging management review results of concrete structures and components may also be used to establish different scheduled inspection frequencies, similar to those recommended by American Concrete Institute 349.3R, for aging management programs. The staff is concerned that the applicant has not proposed to perform periodic inspections of concrete components during the period of extended operation. Therefore, the staff is unable to make a reasonable assurance finding that in-scope concrete structures and components will maintain their structural integrity and intended functions.

Open item 3.5-2. The staff expressed concern that the applicant did not plan to periodically monitor groundwater during the extended period of operation to confirm that it is not aggressive to buried portions of concrete structures. As stated in the applicant's response to RAI 3.5.1, the chloride, sulfate, and pH values over the past 20 to 30 years are well below the limits where potential degradation of concrete may occur. In addition, the water contour tables for both Catawba and McGuire show that the water table levels decrease from the two nuclear stations outward to the surrounding areas such that only a chemical event at the nuclear stations would potentially impact their respective site environments, including the groundwater. However, in its response to RAI 3.5-1, the applicant does not commit to initiate corrective action in the event of a potential change to the site environment resulting from a chemical release during the period of extended operation. Such a corrective action would need to include a commitment to monitor the groundwater chemistry and to assess the potential impact of any changes to the groundwater chemistry on below-grade concrete components.

Open item 3.5-3. Since the ice condenser wear slab, structural concrete floor and crane wall are characterized as inaccessible and in a unique environment of low humidity and temperature, the staff acknowledges that there are no accessible concrete components in a similar environment that the applicant could use as an indicator of the aging of these inaccessible ice condenser components. However, the applicant indicated, in its response to RAI 3.5-6, that portions of both the structural concrete floor, which is located beneath the ice condenser wear slab, and the crane wall are accessible for inspection. Specifically, the applicant stated that the structural concrete floor is accessible from below and that the interior surface of the crane wall is open to the reactor building environment and is accessible for inspection. For the ice condenser wear slab, the applicant did not state in its response that it would inspect the wear slab in the event that defrosting of an ice condenser wall panel allows access to the wear slab. Since the applicant does not plan to inspect potentially accessible portions of the ice condenser crane wall or accessible portions of the ice condenser structural concrete floor, the staff cannot conclude, with reasonable assurance, that these concrete structures will be adequately monitored to ensure that their intended functions will be maintained during the extended period of operation.

<u>New open item 3.5-4.</u> Neither the FSAR Supplement nor the referenced TS and SLCs provide adequate descriptions of the Battery Rack Inspections. The applicant is requested to provide a summary description characterizing the important elements of the Battery Rack Inspections from Section B.3.2 of the LRA and the applicant's response to RAI B.3.2-1.

<u>New open item 3.5-5.</u> The staff reviewed the FSAR Supplement provided in UFSAR Section 18.2.7 as presented in Appendix A-1 and Appendix A-2 of the LRA for McGuire and Catawba, respectively, and compared this information to that which was provided in Section B.3.10 of the LRA and the clarifications provided by the applicant in response to RAI B.3.10-1. Some important industry standards and the NRC guidelines used for the AMP were not incorporated into Section 18.2.7 of the FSAR Supplement. The applicant is requested to update the FSAR Supplement to incorporate the standards and guidelines.

<u>Open item 3.6.1-1.</u> The applicant should provide a technical justification that will demonstrate that visual inspection of high range radiation monitor and high voltage neutron monitoring instrumentation cables will be effective in detecting aging before current leakage can affect instrument loop accuracy.

<u>Open item 4.3-1.</u> In its response to a staff request for pressurizer sub-component cumulative usage factors (CUFs), the applicant indicated that modified operating procedures had been implemented at McGuire and Catawba to mitigate the effects of insurge/outsurge. In addition, historical plant instrument data were analyzed to determine the insurge/outsurge history both before and after modification of the operating procedures. The applicant indicated that an analysis including these events found that the design CUFs of all components will remain less than 1.0. By letter dated July 9, 2002, the applicant provided the CUFs for the sub-components listed in Table 2-10 of WCAP-14574-A but did not discuss the impact of the environmental fatigue correlations on these sub-components. Pending completion of the staff's review of the information provided and assessment of the impact of the environmental correlations for these sub-components, this issue is characterized as an open item.

<u>New open item 4.3-2.</u> By letter dated July 9, 2002, the applicant provided a table of CUFs for newer-vintage Westinghouse plant locations identified in NUREG/CR-6260. The staff's review of these data is ongoing. The Catawba UFSAR lists a large number of design cycles for charging and letdown flow changes. Duke's response to RAI 4.3-5 indicates that these transients cause insignificant fatigue and are not counted. The staff notes that NUREG/CR-6260 contains a discussion of these transients for the newer vintage Westinghouse plant and indicates that these transients are not normally counted at PWRs, although some PWRs have reported that the actual cycles of these transients are less than the numbers assumed in the design calculations. However, the NUREG/CR-6260 evaluation indicates the fatigue usage at the charging nozzle for these transients is significant when the reactor water environment is considered. The charging nozzle is one of the locations Duke will

assess for fatigue environmental effects. As such, Duke should provide the design stresses and fatigue usage factors associated with the Catawba charging system flow changes.

Open item 4.3-3. The staff reviewed the Catawba Updated Final Safety Analysis Report (UFSAR), Section 1.7, Regulatory Guides, and Section 5.3.1.4, Special Controls for Ferritic and Austenitic Stainless Steels, and determined that sufficient information was provided in the UFSAR to conclude that underclad cracking was not a concern for Catawba 1 and 2. The staff also reviewed information, submitted by letter from the applicant dated July 9, 2002, to conclude that underclad cracking is not a concern for McGuire 1. However, the staff does not have sufficient information about the McGuire 2 fabrication process to conclude that underclad cracking is not a concern. If the applicant can not provide conclusive evidence that the fabrication procedure does not result in underclad cracking, then it can furnish an analysis for the license renewal term.

New open item 4.3-4. Duke provided a McGuire FSAR Supplement for Section 3.9.2 and a Catawba FSAR Supplement for Section 3.9.3 which indicates that stress range reduction factors were used in the evaluation of ASME Class 2 and 3 piping systems. Duke also provided a McGuire FSAR Supplement for Section 5.2.1 and a Catawba FSAR Supplement for Section 3.9.1 to indicate that the Thermal Fatigue Management Program (TFMP) will continue to manage thermal fatigue into the period of extended operation. However, Duke did not describe its commitment to evaluate the effects of the environment on fatigue of reactor coolant system pressure boundary components in the UFSAR Supplement. Nor did Duke provide a description of its TFMP. The FSAR Supplement should be revised to reflect this information.

Confirmatory item 2.3.3.26.2-1. By letter dated January 28, 2002, the staff requested, in RAI 2.3.3.26-2, the applicant to indicate if piping and nitrogen cylinders associated with a safety-related backup nitrogen control system were within the scope of license renewal. In its response, dated April 15, 2002, the applicant confirmed that the Catawba main steam line PORVs are supplied with a nitrogen control system backup to the normal instrument air supply. This backup nitrogen control system consists of valves, tubing, and nitrogen bottles. The applicant stated that the nitrogen bottles are periodically replaced and, therefore, are not subject to an AMR. However, the applicant did not specify the details of the periodic replacement. In electronic correspondence dated July 16, 2002, the applicant stated that a Catawba technical specification surveillance procedure requires nitrogen cylinder replacement if the pressure in either nitrogen cylinder is less than or equal to 2420 psig. Pending the staff's receipt of this information in official correspondence, this item is confirmatory.

Confirmatory item 3.6.1-1. The applicant agreed to revise the corrective actions and confirmation process element of the Non-EQ Insulated Cables and Connections Aging Management Program to reflect that the program should consider the potential for moisture in the area of degradation. However, the FSAR supplement needs to be revised to reflect this change to the corrective actions and confirmation process element description.

Confirmatory item 3.6.2-1. The applicant eliminated the qualifier "significant" from its discussion of exposure to moisture. However, the FSAR supplement needs to be revised to reflect this change in the scope of the Inaccessible Non-EQ Medium-Voltage Cables Aging Management Program.

Confirmatory item 4.4-1. To address Generic Safety Issue (GSI) 168, the applicant submitted, in a letter dated July 9, 2002, a technical rationale that demonstrates that the CLB will be maintained until some later point in the period of extended operation, at which time one or more reasonable options would be available to adequately manage the effects of aging. However, the staff requests that the applicant also indicate that it will monitor updates to NUREG-0933, "A Prioritization of Generic Safety Issues," for revisions to GSI-168 during the review of its application, or that it will supplement its license renewal application if the issues associated with GSI-168 become defined such that providing the options or pursuing one of the other approaches described in the SOC becomes feasible.