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 U.S. 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
> Attention: Rani Franovich, Mail Stop O-12D3

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 its 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.

In Section 1.5 of its LRA and in the June 13, 2002, transmittal letter, Duke Energy Corporation made the following request:

> As reflected in these proposed revisions to the license expiration dates, Duke recognizes the legal limits associated with the term of renewed operating licenses. We also note that the technical and environmental reviews performed in connection with this Application cover operation for a period of sixty years. Duke therefore requests that the NRC complete its safety and environmental reviews

such that 60-years of operation are evaluated even though the renewed licenses issued may actually provide somewhat less than an additional 20-years of operation beyond the end of the current operating licenses of one or more of the McGuire or Catawba units.

To accommodate this request, the staff focused its attention on the time-limited aging analyses (TLAAs) provided in Chapter 4 of the LRA and identified the following sections of the LRA that described TLAAs that assumed 60 years of plant operation:

- Section 4.2, "Reactor Vessel Neutron Embrittlement"
- Section 4.3.2, "ASME Section III, Class 2 and 3 Piping Fatigue"
- Section 4.7.1, "Reactor Coolant Pump Flywheel Fatigue"

Other Chapter 4 sections of the LRA identify aging effects that will be managed by an aging management program, in accordance with10 CFR 54.21(c)(iii), or identify aging that is not applicable to either McGuire or Catawba. The staff reviewed the three LRA Sections and associated TLAAs listed above and concluded that they remain valid for 60 years of operation. Therefore, they remain valid for the period of extended operation in accordance with 10 CFR 54.21(c).

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: (1) a technical review of safety issues and, (2) 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 the 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. Appendix D contains a list of commitments provided by the applicant in a letter dated December 16, 2002, and confirmed by the staff.

In accordance with 10 CFR Part 51, the staff prepared draft plant-specific supplements to the generic environmental impact statement (GEIS). The supplements discuss the environmental considerations related to renewing the licenses for McGuire and Catawba. The draft plant-specific supplements to the GEIS were 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 (1) that are safety-related, (2) whose failure could affect safety-related functions, and (3) 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 were issued in December 2002.

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: (1) an integrated plant assessment (IPA), (2) current licensing basis changes during NRC review of the LRA, (3) an evaluation of TLAAs, and (4) 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

As a result of its review, the NRC staff issued an SER with open items on August 14, 2002, and identified and documented 41 open items and 4 confirmatory items. An issue was characterized as an open item if the applicant had not presented a sufficient basis for resolution, or if questions or concerns about the applicant’s license renewal application emerged late in the staff’s review, such that resolution could not be proposed by the applicant before the SER with open items was issued. An issue was characterized as confirmatory if the staff and applicant had agreed to a resolution, but information in official submittals from the applicant was needed. New open items involved issues that had not been the subject of staff RAIs. The applicant responded to the open and confirmatory items, as well as two other emerging issues pertaining to the treatment of electrical fuse holders and aging management of the pressurizer surge and spray nozzle thermal sleeves and the steam generator divider plates, in letters dated October 2, 2002, October 28, 2002, November 5, 2002, November 14, 2002, November 18, 2002, and November 21, 2002. The staff’s evaluation of the applicant’s responses to the emerging issues is documented in Sections 2.5.2.2, 3.1.2.2.1, and 3.6.1.2.1 of this SER.

The applicant’s responses to open and confirmatory items are described below.

<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.

In its response to SER open items 2.3-1 and 2.3-2, dated October 28, 2002, the applicant provided AMR results tables for the fan and damper housings of ventilation systems that are in scope at McGuire and Catawba. The staff found the applicant's response sufficient to resolve open items 2.3-1 and 2.3-2. Because these open items apply to a number of ventilation systems, their resolution is documented in multiple sub-sections of Sections 2.2 and 2.3 of this SER. The staff's evaluation of the AMR results is documented for applicable systems in Sections 3.2 and 3.3 of this SER.

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 was considered inadequate to manage the aging of building sealants, which are long-lived, passive structural sub-components within the scope of license renewal.

In its response to this open item, dated October 28, 2002, the applicant credited a visual inspection of the structural sealant used to maintain ventilation pressure boundary integrity of the control room area, emergency core cooling pump rooms, annulus, and fuel handling building. The staff found the applicant's response sufficient to resolve open item 2.3-3. The staff's evaluation of the Ventilation Area Pressure Boundary Sealants Inspection Program is documented in Section 3.0.3.19 of this SER.

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 hoses in Table 3.3-15, since these components are passive, long-lived, and have intended functions to maintain pressure boundary. 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 was characterized as an open item.

In its response to this open item, dated October 28, 2002, the applicant confirmed that the flexible hose in the diesel generator cooling water system is replaced on a qualified life every 6 years and, therefore, is not subject to an AMR. The staff agreed with this conclusion. Therefore, open item 2.3.3.12.2-1 is closed.

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 was requested either to (1) indicate if replacement of these components is based upon a qualified life or based upon condition or performance monitoring, or (2) specify the parameters that will be monitored as indicators of the components' condition or performance.

In its response to this open item, dated October 28, 2002, the applicant stated that the synthetic rubber flexible hoses on the inlet and outlet of the diesel generator crankcase vacuum blowers are inspected for cracking and signs of wear on a 6-year frequency and replaced based on condition. The staff found this to be an acceptable basis for excluding these hoses from an AMR. Therefore, open item 2.3.3.13.2-1 is closed.

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 was requested either to (1) indicate if replacement of these components is based upon a qualified life or based upon condition or performance monitoring, or (2) specify the parameters that will be monitored as indicators of the components' condition or performance.

In its response to this open item, dated October 28, 2002, the applicant stated that the flexible hoses in the diesel generator fuel oil system are replaced on a qualified life every 6 years and, therefore, are not subject to an AMR. Since the component is replaced on a specified interval, the staff agreed with this conclusion. Therefore, open item 2.3.3.14.2-1 is closed.

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. 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 was requested to 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.

During a meeting with the staff on October 1, 2002, and in its formal response to this open item dated October 28, 2002, the applicant stated that the fire protection plant designs for McGuire and Catawba are unique. By design, most plants rely upon the hydrants for compliance with 10 CFR 50.48 as a backup means of suppression to ensure defense-in-depth. However, the fire protection system in the auxiliary buildings for McGuire and Catawba consists of two headers that feed the automatic and manual suppression systems. These headers provide sectional isolation capability between the automatic and manual suppression systems such that a single failure cannot cause loss of water supply to both the automatic and manual means of suppression in a given area. As such, defense-in-depth exists in the fire protection system design in the auxiliary building for McGuire and Catawba. In addition, Duke stated that no potential sources of radioactive releases are protected in the event of a fire by those hydrants that are excluded from the scope of license renewal at McGuire or Catawba. Since the applicant does not rely on the hydrants as a backup means of suppression or to protect against the release of radioactive releases for compliance to 10 CFR 50.48, SER open item 2.3.3.19-1 is closed.

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 9.5-1 and NFPA 20. The staff was 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.

In its response dated October 28, 2002, Duke identified the jockey pump casings, piping, and other components of the fire water pressure maintenance sub-system as within the scope of license renewal. The applicant also provided the AMR results for the pressure maintenance subsystem of the fire protection system containing the jockey pump. Therefore, the staff was satisfied with the resolution of this issue. Open item 2.3.3.19-2 is closed. The staff's evaluation of the AMR results for the fire water pressure maintenance sub-system is documented in Section 3.3.19.2 of this SER.

<u>Open item 2.3.3.19-3.</u> Duke did not identify Catawba fire suppression equipment that provides fire water to lower containment carbon filters as within the scope of license renewal. Section 9.5.1.2.1 of the UFSAR states that the interior fire water system provides a fixed water suppression system for <u>charcoal</u> 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 <u>carbon</u> 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 (ADAMS Accession No. ML021280003), Duke Specification CNS-1465.00-00-0006 states that <u>carbon</u> filters are protected by built-in water spray systems. The staff did not believe that the applicant's distinction between charcoal and carbon filters was material.

In its response dated October 28, 2002, the applicant stated that it had performed further review and determined that the piping, sprinklers, and valve bodies associated with the Catawba reactor building charcoal filter unit sprinklers should have been identified as within the scope of license renewal and subject to aging management review. The applicant indicated that the components of this portion of the Catawba FP system were listed in Table 3.3-27 of the LRA. Since the fixed water suppression system for the charcoal filters was included in scope and subject to an AMR, the staff was satisfied with its resolution. Open item 2.3.3.19-3 is closed. The staff's evaluation of the AMR results is documented in Section 3.3.19.2 of this SER.

<u>Open item 2.3.3.19-4.</u> 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 the oil storage house, the oxygen and acetylene gas storage yard area, the compressed flammable gas cylinder storage area, the main turbine piping and bearings, the unit start-up and standby oil-filled power transformers, the main turbine lube oil reservoirs, the hydrogen seal oil unit, and the feedwater pump turbines. The UFSARs do not differentiate between those manual hose stations 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 was requested to 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.

In its October 28, 2002, response to this open item, the applicant stated that separation was the only credited fire protection feature for those areas listed in the open item that are located in the yard. The staff agreed with the applicant's finding that the suppression systems in the outlying plant areas did not appear to be credited due to physical separation from surrounding buildings. In an augmented response dated November 18, 2002, the applicant stated that, although it disagreed with the staff's position with respect to manual hose stations in the turbine buildings, the equipment associated with these fire suppression features would be included in the scope of license renewal. The applicant also provided AMR results tables for the passive equipment brought into the scope of license renewal. Therefore, open item 2.3.3.19-4 is resolved. The staff's evaluation of the AMR results is documented in Section 3.3.19.2.

<u>Open item 2.3.3.19-5.</u> The staff agreed 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 was 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). Since the applicant included the pump casings within the scope of license renewal, the staff believed that the strainers also should be within scope.

In its response dated October 28, 2002, the applicant stated that it had performed an AMR for the main fire pump strainers and provided the results of its review. These AMR results for the strainers were generically applicable to both McGuire and Catawba. The applicant indicated that each pump has a strainer that is within the scope of license renewal and is subject to AMR because it is a long-lived, passive component. This staff was satisfied with the resolution of this issue. Open item 2.3.3.19-5 is closed. The staff's evaluation of the AMR results is documented in Section 3.3.19.2 of this SER.

<u>New open Item 2.3.3.19-6.</u> 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 was concerned that the manually operated water spray systems for these filters were inappropriately excluded from the scope of license renewal and an AMR.

In its response dated October 28, 2002, the applicant stated that the flexible hoses, piping, sprinklers, and valve bodies associated with the McGuire reactor building exhaust filters spray system should have been identified as within the scope of license renewal and subject to aging management review. The components of this portion of the McGuire FP system are listed in Table 3.3-26 of the LRA. The staff was satisfied with the resolution of this issue. Open item 2.3.3.19-6 is closed. The staff's evaluation of the AMR results provided in Table 3.3-26 of the LRA is documented in Section 3.3.19.2 of this SER.

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.

In its response to this open item, dated October 28, 2002, the applicant stated that the flexible hoses in the standby shutdown diesel generator fuel oil subsystem are inspected for cracking and signs of wear on an 18-month frequency and replaced based on condition. The staff found this to be an acceptable basis for excluding these hoses from an AMR. Therefore, open item 2.3.3.35.2-1 is closed.

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 was characterized as open.

In its June 26, 2002, letter, the applicant indicated that the following passive component commodity groups (that were originally identified as out of scope) have been identified as being within the scope of license renewal and subject to an AMR: high-voltage insulators, phase bus (e.g., isolated-phase bus, nonsegregated-phase bus, bus duct), switchyard bus, and transmission conductors. In a letter dated October 2, 2002, the applicant clarified its response to SER open item 2.5-1, stating that all insulated cables and connections (power, control, and instrumentation applications) installed in the additional areas identified in the SBO open item response were, and still are, in scope as part of a bounding scope. The applicant also provided, in a letter dated October 28, 2002, a simplified one-line diagram of the SBO power recovery path and further clarified that insulated cables and connections included as part of the SBO power recovery path are considered to be part of the larger component commodity group, which includes all insulated cables and connections. Cables and connections in the SBO power recovery path were considered by the applicant to be within the scope of license renewal and subject to an AMR. Since the long-lived, passive component associated with the offsite power path for recovery from SBO events was included within the scope of license renewal in accordance with 10 CFR 54.4(a)(3), open item 2.5-1 is closed.

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.

In its response dated October 28, 2002, the applicant added references to improved technical specifications (ITS) 5.5.10 and 5.5.13 (for McGuire and Catawba) and SLC requirements (16.5-

7, 16.8-3 and 16.9-7 for McGuire, and 16.5-3, 16.7-9 and 16.8-5 for Catawba). The applicant also augmented its McGuire and Catawba FSAR supplements to indicate that the Chemistry Control Program contains system-specific acceptance criteria that are based on the guidance provided in EPRI PWR Primary Water Chemistry Guidance, EPRI PWR Secondary Water Chemistry Guidelines, and EPRI Closed Cooling Water Chemistry Guidelines. The staff found that the revised FSAR supplement is consistent with the program described in Appendix B of the LRA and considers open item 3.0.3.2.3-1 closed.

<u>New open item 3.0.3.9.1.2(a-g).</u> 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).

In its response to SER open item 3.0.3.9.1.2(a), dated October 28, 2002, the applicant indicated that these heat exchanger tubes are a coil design and, therefore, are not candidates for eddy current testing. As indicated in Section B.3.17.6 of the LRA, either destructive or non-destructive examination will be performed to examine the internal surfaces of the tubes. If evidence of loss of material is observed during the initial inspection, a problem report will be initiated in accordance with the problem investigation process defined in Nuclear System Directive 208. The problem investigation process is a formalized process for documenting engineering evaluations of plant problems that would include the assessment of the severity of the observed degradation, the need for corrective actions, the need for further inspections of other locations, and the need for future inspections or programmatic oversight. Criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions. Any criteria or analysis methods involved in determining the severity of the degradation and the need for corrective action will be developed at the time of the evaluation and will be a part of the problem report. Since the applicant indicated that it would consider the ASME Code (which is endorsed by the staff through 10 CFR 50.55a) and other pertinent factors in determining the acceptance criteria for loss of material, the staff found the applicant's response to SER open item acceptable. Therefore, open item 3.0.3.9.1.2 (a) is closed.

In its response to SER open item 3.0.3.9.1.2(b-g), dated October 28, 2002, the applicant indicated that criteria such as ASME Code requirements, additional inspection results, and operating experience may be used to assess the severity of the degradation and the need for corrective actions. The applicant further explained that eddy current testing at McGuire and Catawba is performed by a vendor who specializes in the practice, and that a four-step process is used to determine if test results are acceptable and generate the final test report. This process was described in detail in the applicant's October 28, 2002, response to this SER open item. The staff found that appropriate and adequate acceptance criteria for detecting heat exchanger tube degradation from loss of material were identified for these aging management programs. Therefore, open items 3.0.3.9.1.2 (b-g) are closed.

<u>New open item 3.0.3.10.2-1.</u> Since 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 staff believed that volumetric examination of a sample of small-bore Class-1 piping was needed to demonstrate that the effects of aging are being adequately managed during the period of extended operation. The staff also believed

that a sample of affected welds selected for inspection should be based upon piping geometry, pipe size and flow conditions, and that the inspection should be performed by qualified personnel using approved station procedures.

In its response dated November 14, 2002, the applicant stated a set of susceptible small bore piping locations will be volumetrically examined on each unit. Locations to be examined will be determined based on consideration of damage mechanisms. Damage mechanisms to be considered include fatigue, stress corrosion, and flow assisted corrosion/flow wastage. Cracking due to thermal fatigue resulting from stratification of fluids and turbulent penetration flow is an aging effect that also will be addressed. The applicant further indicated that the Small Bore Piping Examination will be an activity within the Inservice Inspection Plan during the period of extended operation. Small Bore Piping Examinations will be performed during each inservice inspection interval during the period of extended operation. By letter dated November 21, 2002, the applicant augmented its response to clarify how the Small Bore Piping Examination will be implemented at McGuire and Catawba. The applicant stated that it will first determine the population of Duke Class A piping that is less than 4-inch NPS for the unit to be inspected. This population of piping will then be reviewed by experienced engineers to determine the more likely locations that could be impacted by the various damage mechanisms described in Duke's November 14, 2002, response to this open item. The determination will involve a review of the physical plant design such as piping layout, geometry and operating temperatures, as well as both plant and industry operating experience that could indicate more optimum inspection locations. The set of locations selected will comprise the scope of the Small Bore Piping Examination and will be identified within the Inservice Inspection Plan for each station. Since volumetric inspection will ensure that the inspections of the small bore piping components will be capable of detecting cracking in the components, the staff considers SER open item 3.0.3.10.2-1 closed.

<u>New open item 3.0.3.10.2-2.</u> 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 requested 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 requested the applicant to describe the actions it plans to take to address this operating experience as it applies to McGuire and Catawba.

In its response to open item 3.0.3.10.2-2, dated October 28, 2002, the applicant specified the McGuire and Catawba reactor coolant system piping that contains welds fabricated from Alloy 82/182 material, and the applicant described the actions it has taken, and will take in the future, to address this operating experience as it applies to McGuire and Catawba. The applicant further stated that the applicable V.C. Summer hot leg safe-end weld was fabricated using a field weld process and was not machined to a smooth bore nozzle configuration as was the case for the corresponding welds at McGuire 1 and 2 and Catawba 1 and 2. The applicant stated that UT examination methods cannot provide accurate results when good contact is not maintained between the UT probe and the weld surface during the examination. The applicant stated that the irregular weld surface at V.C. Summer was the contributing factor for the inability of the UT inspections to provide relevant inspection results. In contrast, the applicant noted that the corresponding welds at McGuire and Catawba were machined to smooth surfaces.

The staff notes that, although the smooth surfaces for McGuire and Catawba welds, described in the applicant's response, may improve the quality of UT examinations, they alone do not ensure that completely accurate, reliable UT examination results can be obtained. The staff is also currently assessing whether the automated UT inspection techniques developed by the EPRI Materials Reliability Project (MRP) Alloy 600 ITG, Alloy 82/182 Weld Integrity Inspection Committee (including those developed by Framatome Technologies, Inc., on behalf of the Alloy 82/182 Weld Integrity Committee) are acceptable methods for detecting PWSCC in RCS hot-leg nozzle safe-end welds fabricated from Alloy 82/182 weld materials. Therefore, the staff still considers PWSCC of the weld material to be a potential aging effect for the McGuire and Catawba RCS pipe welds identified in the applicant's response to SER open item 3.0.3.10.2-2.

The staff is assessing the generic applicability of this current operating issue and is pursuing its resolution pursuant to 10 CFR Part 50. Any required activities associated with its resolution (still under review) will be implemented by the applicant during the current operating term to ensure that the integrity of the Class 1 safe-end welds will be maintained consistent with the CLB before the period of extended operation begins. Thus, pursuant to 10 CFR 54.30, the V.C. Summer issue, as it relates to the structural integrity of the McGuire and Catawba hot-leg nozzle safe-end welds, is outside the scope of the license renewal review. Since the applicant provided the information requested in SER open item 3.0.3.10.2-2 (locations of 82/182 weld material in the RCS piping and activities to address the V.C. Summer operating experience), and since, pursuant to 10 CFR 54.30, the V.C. Summer hot leg cracking event is beyond the scope of the staff's license renewal review, open item 3.0.3.10.2-2 is closed.

<u>New open item 3.0.3.11.3-1.</u> The FSAR supplements did not include references 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 staff requested the applicant to submit an updated summary description of the program to reflect these codes and standards.

In its response dated October 2, 2002, the applicant provided an update of the FSAR supplements for McGuire and Catawba. These updates included references to NRC Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," and ACI 349.3, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," which were included in the applicant's response to RAI B.3.21-2. Therefore, open item 3.0.3.11.3-1 is closed.

<u>New open item 3.0.3.13.2-1.</u> 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 believed 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 the most vulnerable locations.

After the SER with open items was issued, the staff reconsidered its assessment of the proposed program. In an electronic correspondence dated September 23, 2002, the staff notified the applicant that open item 3.0.3.13.2-1 was considered resolved. Corrosion of the outside surface of a buried pipe occurs at locations where the coating is damaged. Since this can happen anywhere along the pipe, the whole length of the pipe would need to be excavated to obtain meaningful information. However, this is not practical. If a leak develops due to corrosion of the outside of a pipe (due to damage of the outside coating), the inside coating would also exhibit signs of damage. Therefore, inspection of the inside coating will reveal the location of the leak. The degree of degradation of the inside coating can give some idea of the condition of the outside coating. Since the sample of internal pipe at McGuire and Catawba to be inspected consists of about 90 percent of the population of piping governed by the Condenser Circulating Water System Internal Coating Inspection program, this significant sample size should yield valid, reliable results with a high degree of confidence. Additionally, the staff found a similar inspection program for Oconee acceptable. Therefore, open item 3.0.3.13.2-1 is considered closed.

<u>New open item 3.0.3.15.2-1.</u> 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's intended function, since sufficient flow at prescribed pressures can still be provided by the system. The applicant also stated 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 believed that localized corrosion can result in the loss of the intended function to maintain pressure boundary under a design basis event before the corrosion reveals itself as pinhole leaks. Therefore, the applicant was requested to 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).

In its response dated October 28, 2002, the applicant provided a more detailed description of its program for inspecting piping in the service water system. The program utilizes ultrasonic technology to look for loss of material. The periodic ultrasonic testing (UT) identifies any potential areas of severe degradation by corrosion that could exceed the ability of piping to maintain its structural integrity. Although the primary issue addressed by the program is gross wall loss, which could lead to structural instability, the program also includes the areas containing localized corrosion by pitting and other localized corrosion mechanisms. This was required because localized corrosion may become a structural concern when a significant number of pinholes are present in a one area. When an occurrence of localized corrosion is identified either by UT or a pinhole leak, an evaluation is performed to justify structural integrity of the inspected component under all design conditions. This ensures that the service water corrosion program addresses localized corrosion affecting structural integrity of the affected components before it is revealed as a pinhole leak. In order to achieve this, the program was designed to perform appropriate inspections, evaluations, and trending and taking appropriate corrective actions. The staff found that, by following this process, the applicant will be able to detect the effects of localized corrosion from pitting and MIC before structural integrity of the piping is jeopardized. Therefore, open item 3.0.3.15.2-1 is closed.

<u>New open item 3.0.3.18.3-1.</u> The FSAR supplements did not include references to some important industry standards and the NRC guidelines used for the Underwater Inspection of

Nuclear Service Water Structures program. The staff requested that the applicant revise its FSAR supplements for McGuire and Catawba to reflect these standards and guidelines.

In its response dated October 2, 2002, the applicant provided a revised FSAR supplement that included the appropriate industry standards. The staff found that the revised FSAR supplement provides a summary description of the program at a level of detail commensurate with that which is provided in the staff's review guidance (Appendix A of NUREG-1800) and is, therefore, acceptable. Therefore, open item 3.0.3.18.3-1 is resolved.

<u>New open item 3.1.2.2.2-1.</u> 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 requested 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 item included the potential need to revise the acceptance criteria for this program and the FSAR supplement summary description.

In its response to open item 3.1.2.2.2.-1, dated October 28, 2002, the applicant stated that the visual inspection method for the pressurizer spray head examination will be revised to VT-1 examination methods, and that the acceptance criteria will be in accordance with those specified for VT-1 examinations in Section XI of the ASME Boiler and Pressure Vessel Code. The applicant also stated that these changes will be reflected in a revision of the UFSAR supplement. The applicant's response indicated that the applicant will implement a visual examination method for the pressurizer spray head examination that is capable of detecting surface cracks in the spray head material, and that any cracks detected by the examination will be evaluated using established Section XI acceptance criteria. This meets the criteria in Section XI of the ASME Code for performing visual examinations of Code Class components for cracking and resolves the issue raised in open item 3.1.2.2.2-1. Therefore, the staff considers open item 3.1.2.2.2.-1 to be closed.

<u>New open item 3.1.3.2.2-1.</u> 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 standby capsule and would be withdrawn at a fluence that is significantly above the equivalent of 60 years. The staff was concerned that the applicant would need 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, the staff was concerned that capsule "U" (a standby capsule) would need to be inserted in the reactor vessel and begin to accumulate fluences in an operating environment for data collection purposes. The staff believed that the applicant should place all pulled capsules in storage so that they may be saved for future use. In addition, the staff believed that, after the applicant has pulled all the capsules, it should use alternative dosimetry to monitor neutron fluence during the period of extended operation. The staff requested the applicant to describe its plans for this capsule.

In its response to open item 3.1.3.2.2-1, dated October 28, 2002, the applicant identified those surveillance capsules that are in storage and those that are available for further testing if necessary. The applicant discussed its RV material surveillance programs for McGuire and

Catawba and clarified its plans for removal and testing of surveillance capsule W (for McGuire 1) and surveillance capsule U (for Catawba 2). The staff concluded that the surveillance program is acceptable for the period of extended operation for all units and considers open item 3.1.3.2.2-1 closed.

<u>New open item 3.1.3.2.2-2.</u> The staff and nuclear power industry are pursuing resolution of the reactor vessel penetration nozzle cracking issue and the Davis Besse 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, PWR upper RV heads, and other RCS piping and components (specifically with respect to non-destructive examinations and the ability to detect cracking in the VHP nozzles and loss of material due to boric acid corrosion). These are emerging, current license issues that have not yet been resolved and, pursuant to 10 CFR 54.30(b), are beyond the scope of this license renewal review. However, since these issues might not be resolved prior to issuance of the renewed operating licenses for the McGuire and Catawba units, the staff requested the applicant to commit to implementing any actions, as part of the VHP Nozzle Program, that are agreed upon between the NRC, the NEI, Materials Reliability Project (MRP), and the nuclear power industry to monitor for, detect, evaluate, and correct cracking in 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.

In its response dated October 28, 2002, the applicant provided revised FSAR supplement summary descriptions of the VHP Nozzle Program and the Alloy 600 Review to indicate that these programs will be revised as necessary to reflect any new or revised commitments made by Duke in response to staff generic communications. The commitment to incorporate resolution of this current operating issue into the VHP Nozzle Program and the Alloy 600 Review, as stated in the revised FSAR supplements, ensures that the methods implemented by the applicant for inspecting the McGuire and Catawba VHP nozzles and RV heads will be sufficient to detect PWSCC in the VHP nozzles. Therefore, the staff found that there was reasonable assurance that the applicant has demonstrated that the effects of aging associated with the VHP Nozzle Program and the Alloy 600 Review will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). The staff considers open item 3.1.3.2.2-2 closed. With respect to boric acid corrosion, the staff is continuing to gather information on industry programs to determine what, if any, regulatory action is needed.

<u>New open item 3.1.4-1(a).</u> 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 believed that the applicant should schedule inspections 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.

In its response to open item 3.1.4-1(a), dated October 28, 2002, the applicant clarified that all of the RV internals for the McGuire and Catawba units were manufactured by Westinghouse, not by the fabricators of the RVs (i.e., neither Combustion Engineering nor Rotterdam Drydock fabricated the RV internals). The applicant provided an acceptable design-feature-based argument for concluding the baffle bolts and plates at McGuire were limiting in regard to the temperatures and fluences the materials would achieve when compared to those in the Catawba units, and stated that it would inspect the RV internals at both McGuire 1 and McGuire 2 during the periods of extended operation for the units and to use the results of the examinations as the basis for determining whether additional inspections of the RV internals at Catawba 1 and Catawba 2 would be necessary. The applicant stated that the RV internals at McGuire 1 will be inspected during the fifth ISI interval for the unit, and the RV internals at McGuire 2 will be inspected during the sixth ISI interval for the unit. Based on this response, the applicant will be performing inspections of the RV internals at five of the seven Duke-owned nuclear reactors (i.e., at Oconee and McGuire). Since the McGuire RV internals are projected to be limiting in comparison to those at Catawba, the staff concluded that the applicant's credited inspections for the RV internal core barrel components at McGuire (and at Oconee) will provide an acceptable basis for determining whether age-related degradation is applicable in the corresponding components at Catawba and for scheduling inspections at Catawba as necessary. This resolves open item 3.1.4-1(a).

<u>New open item 3.1.4-1(b).</u> The critical crack size acceptance criterion for RV internal forgings, plates, and welds, and RV internals made from CASS had not yet been established. Nor had 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. The staff requested a commitment from the staff to determine the critical crack size and submit this acceptance criterion (when it has been determined) to the staff.

In its response to open item 3.1.4-1(b), dated October 28, 2002, the applicant provided a summary description of the Acceptance Criteria attribute of the Reactor Vessel Internals Inspection for each station's FSAR supplement to address the need to submit the acceptance criteria established by industry programs for evaluating cracking, loss of fracture toughness, and void swelling in Westinghouse-designed RV internals to the staff for review and approval. This is acceptable to the staff, since the industry is currently in the progress of establishing what the techniques and acceptance criteria will be for evaluation of these aging effects in Westinghouse-designed RV internals. This resolves open item 3.1.4-1(b).

<u>New open item 3.1.4-1(c).</u> The staff requested the applicant to provide a commitment to update the "Detection of Aging Effects" program attribute in FSAR Supplement Section 18.2.22, "Reactor Vessel Internals Inspection," to reflect the second paragraph in the applicant's response to RAI B.27-2. This part of open item 3.1.4-1 was not identified in the SER with open items. For tracking purposes, the staff and applicant characterized this issue as SER open item 3.1.4-1(c).

In its response to open item 3.1.4-1(c), dated October 28, 2002, the applicant stated that the FSAR supplements for McGuire and Catawba will be revised to incorporate a statement that the visual inspection method selected for the inspection of RV internal plates, forging, and welds will be sufficient to detect cracks in the components prior to any growth to a size that is greater than the critical crack size (critical crack length) for the material. In its response, the applicant acknowledged that, for visual inspections of RV internals at McGuire and Catawba, it must implement a visual inspection technique that is capable of detecting surface cracks in the internal components. This acknowledgment resolves open item 3.1.4-1(c).

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

In its response dated October 28, 2002, the applicant provided a modified FSAR supplement summary description of this program. The revised FSAR supplement included a statement that inspections of the steam generator surveillance program follow the recommendations of NEI 97-06, “Steam Generator Program Guidelines.” The staff found the changes acceptable because the modified FSAR supplement summary description will be consistent with the steam generator surveillance program described in Appendix B, Section B.3.31, of the Catawba and McGuire LRA. The staff considers open item 3.1.5-1 closed.

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 was requested to submit a more detailed description of the yard environment for the condenser circulating water system expansion joints to address UV exposure.

In its response dated November 14, 2002, the applicant agreed to add cracking and wear as potential aging effects and addressed the issue of potential degradation of the synthetic rubber expansion joint in the condenser circulating water system. The applicant stated that it would implement a one-time inspection of the expansion joints in order to characterize any cracking and wear of expansion joints exposed to raw water internal and the yard external environments. The applicant stated that, based on current operating experience, one-time inspection of the expansion joints will be adequate for protecting the system. The staff reviewed the AMR results and concluded that the aging effects specified for the expansion joint were consistent with industry experience for these combinations of materials and environments. The staff also evaluated the one-time inspection credited for these components and found that there was reasonable assurance that the applicant had demonstrated that the effects of aging associated with the one-time inspection of the expansion joints in the condenser circulating water system program will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21(a)(3). Therefore, the staff considers open item 3.3.6.2.1-1 resolved.

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 was requested 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 to identify this aging effect and an AMP to manage or monitor the associated loss of material.

In its response dated October 28, 2002, the applicant provided a revised AMR results table for the diesel generator starting air distributor filter. The applicant specified loss of material as an aging effect and credited the Inspection Program for Civil Engineering Structures and Components. The aging effect specified is consistent with industry experience for the material and environment specified. Therefore, this response is acceptable to the staff and resolves open item 3.3.17.2.1-1.

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 did 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 recognized that sufficient flow through the heat exchanger may prevent areas of stagnation in which fouling may occur. However, the applicant had not substantiated its conclusion with any operating experience, such as maintenance and surveillance results, to demonstrate the success of this activity in preventing fouling. With respect to the filtering of the treated water to remove particles, the staff recognized 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.

In its response dated October 28, 2002, the applicant identified fouling due to silting as an aging effect requiring management for the heat exchanger in the standby shutdown diesel cooling water and jacket water heating subsystem. The applicant further clarified that the standby shutdown diesel cooling water and jacket water heating subsystems are closed cooling water systems treated with corrosion inhibitors. The Chemistry Control Program was credited for managing fouling. The staff found that the clarifications and changes provided by the applicant are appropriate to ensure that the aging effects associated with the heat exchanger in the standby shutdown diesel cooling water and jacket water heating subsystem will be adequately managed during the period of extended operation. The identification of fouling as an aging effect and its management through corrosion inhibitors monitored by the Chemistry

Control Program were acceptable because the program precludes the formation of corrosion products that can cause the fouling of the heat exchanger and adversely impact the heat transfer function. Therefore, open item 3.3.35.2-1 is closed.

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

In its response dated October 28, 2002, the applicant stated that it had searched the operating experience database to determine if there had been any component failures, relevant industry operating experience, or problems discovered during routine maintenance and testing. The applicant did not find any loss of the intended functions of the auxiliary feedwater system components that could be attributed to the inadequacy of the chemistry control program. The applicant stated that routine maintenance of other secondary system components, such as the steam generators and main turbine, provides additional operating experience because they do operate during startup and shutdown and are of the same chemistry as the feedwater system and other secondary side systems. These secondary systems have also shown no degradation affected by water chemistry. However, the applicant added a statement to Section 18.3 of the McGuire and Catawba FSAR supplements to indicate that visual inspections of the interior surfaces of auxiliary feedwater system and main feedwater system components and piping will be performed when available, and that the inspection results will be documented in writing and available for inspection following issuance of renewed operating licenses for McGuire and Catawba. The staff finds the augmented Catawba and McGuire FSAR supplements acceptable because the applicant will inspect these internal surfaces specifically for aging effects (loss of material) and will document its findings in the inspection procedure. This deliberate inspection will provide an opportunity to verify that the Chemistry Control Program is effective and thereby satisfies the intent of the one-time inspection. The staff considers open item 3.4.1.2.2-1 closed.

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 staff believed the applicant should periodically inspect these components. Although the applicant had 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's position (issued by letters dated November 23, 2001, and April 5, 2002, 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 was concerned that the applicant had not proposed periodic inspections of concrete components during the period of extended operation. Therefore, the staff was unable to make a reasonable assurance finding that in-scope concrete structures and components would maintain their structural integrity and intended functions.

In its response dated October 2, 2002, the applicant agreed to resolve open item 3.5-1 by committing to manage the aging of accessible concrete structural components during the period of extended operation. In a letter dated October 28, 2002, the applicant submitted revised AMR results tables for Section 3.5 of its LRA. In a letter dated November 14, 2002, the applicant state that it would manage loss of material, cracking, and change in material properties for the accessible concrete components identified in Tables 3.5-1 and 3.5-2 of the LRA. The applicant credited the Inspection Program for Civil Engineering Structures and Components to manage the specified aging effects. The applicant's periodic inspection of accessible concrete structures and components through its Inspection Program for Civil Engineering Structures and Components is acceptable to the staff. Therefore, open item 3.5-1 is closed.

<u>Open item 3.5-2.</u> 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 did 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.

In a letter dated July 9, 2002, the applicant stated that it did not commit to initiate a corrective action in the event of a potential change to the site environment resulting from a chemical release during the period of extended operation, because such an event was not postulated. The applicant stated that it was not credible to postulate that some environmental event will occur in the future that would affect the quality of groundwater in the vicinity of Catawba or McGuire. A change in the environment due to a chemical release would be an abnormal event. The staff reviewed NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," and determined that aging effects from abnormal events need not be postulated specifically for license renewal. After the SER was issued with this identified as open item 3.5-2, the staff reviewed the guidance provided in NUREG-1800 and reconsidered the applicant's assertion that a potential change to the site environment resulting from a chemical release during the period of extended operation would be an abnormal event. The staff agreed that such a chemical release would not need to be postulated for the purposes of performing an aging management review for license renewal. Therefore, the staff closed open item 3.5-2 without any further information from the applicant. The applicant was notified of this resolution by electronic correspondence dated September 3, 2002.

<u>Open item 3.5-3.</u> Since the ice condenser wear slab, structural concrete floor, and crane wall were characterized as inaccessible and in a unique environment of low humidity and temperature, the staff acknowledged 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 accessible for

inspection. For the ice condenser wear slab, the applicant indicated that a protective layer of ice would prevent water from coming into contact with the wear slab. Since the applicant did 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 could not conclude, with reasonable assurance, that these concrete structures would be adequately monitored to ensure that their intended functions will be maintained during the extended period of operation.

In its response to open item 3.5-3, dated October 2, 2002, the applicant stated it had performed an additional review of the design of McGuire and Catawba and determined that the ice condenser wear slab was not within the scope of license renewal because it did not perform a license renewal function. With respect to the other structures identified in the SER open item, the applicant stated that it disagreed with the staff's conclusion that these structural components require aging management for the period of extended operation. Nonetheless, the applicant stated that it would perform periodic inspections of the accessible portions of the crane wall and ice condenser structural concrete floor during the period of extended operation as part of the Inspection Program for Civil Engineering Structures and Components. Since the ice condenser wear slab does not perform an intended function that meets the license renewal scoping criteria specified in 10 CFR 54.4, the staff agrees with the applicant's finding that the wear slab should not have been included within the scope of license renewal. The staff's review of this item is documented in Section 2.4.1.3.2 of this SER. In addition, since the applicant stated that it would manage the aging effects for the accessible portions of the crane wall and ice condenser structural concrete floor during the period of extended operation (as indicated in its response to SER open item 3.5-1), the staff considers open item 3.5-3 to be closed.

<u>New open item 3.5-4.</u> Neither the FSAR supplement nor the referenced TS and SLCs provided adequate descriptions of the Battery Rack Inspections. The applicant was 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.

In its response dated October 2, 2002, the applicant provided a revision to Table 18-1 and Section 18.3 of the FSAR supplements for McGuire and Catawba. The revised FSAR supplements specified that inspections of the structural supports and anchorages of the battery racks would be performed. The staff found the applicant's revisions acceptable, since inspection of these specific sub-components of the battery rack structures was specified. Open item 3.5-4 is considered closed.

<u>New open item 3.5-5.</u> The staff reviewed the FSAR supplement provided in Appendix A-1 and Appendix A-2 of the LRA for McGuire and Catawba, respectively, and compared this information to that 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 was requested to update the FSAR supplements to incorporate the standards and guidelines.

In its response dated October 2, 2002, the applicant submitted revised McGuire and Catawba summary descriptions of the Monitoring and Trending attribute for this inspection program, which incorporated reference to the codes and standards listed in the RAI response. The staff found the applicant's revision to the FSAR supplements acceptable because the revisions ensure that the program will be governed by these codes and standards. Therefore, open item 3.5-5 is closed.

Open item 3.6.1-1. The applicant was requested to provide a technical justification that would demonstrate that visual inspection of high range radiation monitor and high voltage neutron monitoring instrumentation cables would be effective in detecting aging before current leakage could affect instrument loop accuracy.

In its response to open item 3.6.1-1, dated October 2, 2002, the applicant reiterated its view that visual inspections have proven to be effective and useful because visual inspections have revealed potential problems. In a subsequent response dated November 14, 2002, the applicant stated that it will implement a program specifically to resolve open Item 3.6.1-1. The name of this program is the License Renewal Program for Non-EQ Neutron Flux Instrumentation Circuits. The scope of this program includes only non-EQ neutron flux instrumentation cables that are within the scope of license renewal. The other cables under discussion here, high-range radiation monitors/cables and the wide-range neutron flux monitors/cables, are included in the McGuire and Catawba EQ program and already covered for license renewal by this program. The staff found the applicant's response to SER open item 3.6.1-1 acceptable because the applicant will implement an AMP to monitor the aging of these sensitive cables. The staff also determined that the program established reasonable assurance that the intended function of electrical cables that are (1) not subject to the EQ requirement of 10 CFR 50.49, and (2) used in circuits with sensitive, low-level signals exposed to adverse localized environments caused by heat, radiation, or moisture will be maintained consistent with the CLB through the period of extended operation. Therefore, open item 3.6.1-1 is closed.

New open item 4.2-1 (not identified in the SER with open items). By letter dated September 13, 2002, the staff requested additional information regarding the impact of the fracture toughness data from the Diablo Canyon 2 surveillance capsule on the PTS assessments for the longitudinal RV beltline welds fabricated from heat No. 21935/12002 at the end of the extended operating term (or end of life extended or EOLE). For tracking purposes, this request was characterized by the staff as open item 4.2-1.

In its response to open item 4.2-1, dated October 28, 2002, the applicant provided revised PTS and USE evaluations for these welds. The staff independently assessed the applicant's response to open item 4.2-1 and revised PTS and USE evaluations for the McGuire 1 RV welds and concluded that the revised RT_{PTS} value for these welds at end of life extended meets the screening criterion for longitudinal welds as stated in the PTS rule and demonstrates that the McGuire 1 RV will comply with the fracture toughness and PTS criteria of 10 CFR 50.61 through the end of the extended period of operation for McGuire 1.

The staff also concluded that the revised USE value for applicable welds at EOLE is above 50 ft-lb screening criterion of the rule for ferritic materials in the irradiated condition and demonstrates that the McGuire 1 RV will comply with the USE screening criteria of 10 CFR Part 50, Appendix G, Section IV.A.1, through the expiration of the extended period of operation for McGuire 1. Therefore, the staff concludes that the applicant's TLAA for the PTS and USE evaluations of McGuire 1 are acceptable pursuant to 10 CFR 54.21(c)(1)(ii). This resolves open item 4.2-1.

Open item 4.3-1. 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 was characterized as an open item.

In its letter dated July 9, 2002, the applicant identified several pressurizer sub-components with relatively high design CUFs for McGuire and Catawba. These sub-components include the shell, spray nozzle, lower head heater penetration and nozzle weld, instrument nozzle, and surge nozzle. An assessment by the staff applying a conservative estimate of the environmental factor to these locations indicated that the CUFs may exceed 1.0 during the period of extended operation. However, Turkey Point and North Anna/Surry license renewal applicants used a combination of quantitative and qualitative assessments to argue that the actual CUFs, including environmental effects, are not expected to exceed 1.0 during the period of extended operation. If similar quantitative and qualitative assessments were performed for McGuire and Catawba, the staff would expect similar results to be obtained because McGuire and Catawba are Westinghouse NSSS designs, like Turkey Point, North Anna and Surry. The applicant stated that it would perform further evaluation of the surge line nozzle during the period of extended operation. The staff concludes that the applicant can use the surge line nozzle evaluation as a representative sample to address environmental effects on pressurizer sub-components for McGuire and Catawba during the period of extended operation. If the further evaluation of the surge line identifies the need for additional actions during the period of extended operation, then the applicant should demonstrate the acceptability of pressurizer sub-components, considering environmental fatigue effects, as part of its corrective action. The staff considers open item 4.3-1 closed.

New open item 4.3-2. 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.

In its response dated October 2, 2002, the applicant discussed the Catawba charging system flow transients. The applicant indicated that a review of the existing engineering calculations found that the charging and letdown flow change transients cause insignificant fatigue usage.

The staff also reviewed the engineering calculations during a September 18, 2002, meeting with the applicant (summarized by memorandum dated November 18, 2002) and confirmed that the Catawba charging flow transients were determined to cause insignificant fatigue usage. In its July, 9, 2002, submittal, the applicant identified relatively high design basis fatigue usage factors for the RPV outlet nozzle, surge line hot leg nozzle, charging nozzle, and safety injection nozzle for McGuire and Catawba. An assessment by the staff, applying a conservative estimate of the environmental factor to these locations, indicated that the CUFs of these components may exceed 1.0 during the period of extended operation. The applicant stated that it would perform further evaluations of these components, considering environmental effects, prior to the period of extended operation in response to SER open item 4.3-4. This commitment is provided in the revised FSAR supplements for Catawba and McGuire submitted by the applicant in a letter dated October 2, 2002. Therefore, open item 4.3-2 is closed.

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 cannot provide conclusive evidence that the fabrication procedure does not result in underclad cracking, then it can furnish an analysis for the license renewal term.

In its response dated October 28, 2002, the applicant stated that Duke had compared the number of design cycles and transients used in the analysis contained in WCAP-15338 with the applicable number of design cycles and transients contained in McGuire Unit 2 design documents, and verifies that WCAP-15338 bounds the number of operating cycles and transients not only for McGuire 2, but also for Catawba Unit 1, whose RV is also fabricated from A508 Class 2 forging segments. In its response to open item 4.3-3, the applicant provided an FSAR supplement summary description to reflect that fatigue analysis in WCAP-15338 for RV underclad cracks in Westinghouse-designed reactors was bounding for the evaluation for RV underclad cracks at McGuire 2. Since the conclusions in WCAP-15338 are bounding and applicable to the evaluation of fatigue-induced crack growth of underclad cracks in the McGuire 2 RV, the staff concludes that the applicant has demonstrated that its analysis for postulated underclad cracks in the McGuire 2 RV remains valid for the extended operating period for McGuire 2, and that the applicant’s TLAA for RV underclad cracks at McGuire 2 is acceptable pursuant to 10 CFR 54.21(c)(1)(i). The staff considers SER open item 4.3-3 closed.

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 indicate 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 FSAR supplement. Nor did Duke provide a description of its TFMP. A revised FSAR supplement was requested to reflect this information.

In its response dated October 28, 2002, the applicant provided FSAR supplements for Catawba and McGuire. The revised FSAR supplements provided summary descriptions of the TFMP for McGuire and Catawba. The revised FSAR supplements also included the applicant's commitment to perform additional evaluations of the effects of environmental fatigue on the critical locations identified in NUREG/CR-6260 prior to the period of extended operation. Therefore, open item 4.3-4 is closed.

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 was characterized as confirmatory.

In its response to this confirmatory item, dated October 28, 2002, the applicant formally provided the information that had been furnished in electronic correspondence. The staff finds that the response provides an acceptable basis for excluding these nitrogen bottles from an AMR. Therefore, confirmatory item 2.3.3.26.2-1 is closed.

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 needed to be revised to reflect this change to the corrective actions and confirmation process element description.

In its response dated October 2, 2002, the applicant stated that it will add a statement to the Corrective Action & Confirmation Process of the Non-EQ Insulated Cables and Connections Aging Management program summary description contained in Chapter 18 of each station's FSAR supplement to indicate that corrective action should consider the potential for moisture in the area of degradation. The staff found the applicant's response to confirmatory item3.6.1-1 acceptable because the modification to the Non-EQ Insulated Cable and Connections Aging Management Program is reflected in the revised FSAR supplement. Confirmatory item 3.6.1-1 is closed.

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.

In its response dated October 2, 2002, the applicant stated that it will insert the summary description of the revised Inaccessible Non-EQ Medium Voltage Cables AMP (as provided in Duke letters dated July 9, 2002, Attachment 1, pages 89-91, and November 5, 2002) in each station's FSAR supplement in place of the program description previously provided. The staff found the applicant's response to confirmatory item 3.6.2-1 acceptable because the change to the program provided by the applicant will be reflected in the FSAR supplement.

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 requested 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.

In its response dated October 2, 2002, the applicant stated that, if the staff were to issue a generic communication that defines the issues associated with GSI-168 such that providing the options or pursuing one of the other approaches described in the SOC to 10 CFR 54 (FR Vol.60, No.88, May 8,1995) becomes feasible, then Duke would supplement its license renewal application. However, the applicant also specified that the staff generic communication should be issued prior to November 1, 2002, in order for Duke to evaluate its contents, prepare a response as a current licensing basis change, if any is required, and provide a supplement to the application (if necessary) in sufficient time for the staff to complete its review prior to the scheduled issuance of the SER for license renewal on January 6, 2003. The resolution to GSI-168 was not issued by the staff prior to November 1, 2002; thus, the applicant's alternative commitment is their original commitment that was stated above in their June 17, 2002, response to GSI-168. Pursuant to the requirements of Part 50, the staff will evaluate the applicant's compliance to the resolution of GSI-168 after its issuance and prior to the extended period of license renewal as part of 10 CFR 50.49 time-limited aging analyses. Resolution of GSI-168 pursuant with Part 50 meets the requirement of 10 CFR 54.21(c)(1)(iii) and is therefore considered acceptable. Confirmatory item 4.4-1 is considered closed.