Informal Response to McGuire Units 1 & 2 and Catawba Units 1 & 2 Safety Evaluation Report with Open Items

* Electrical Related Items *

09/17/2002

Enclosure 3

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

By electronic communication on September 4, 2002, the NRC staff provided the following:

> EEIB has reviewed the June 26, 2002, Duke response which provides the results of the AMR for structures and components relied upon to restore power from offsite sources following station blackout. While transmission conductors were included in the scope and subject to an AMR, the staff noted that the aging management review results did not include any insulated control or power cables associated with the switchyard, transformer station, and relay house. The submittal did however, identify switchyard cable trenches, and transformer station cable trenches, providing routing, support and protection to cables servicing the switchyard and relay house equipment. This item remains open pending the determination of whether the insulated power and control cables associated with the switchyard, transformer station, and relay house are in scope for SBO recovery.

Duke Response to Open Item 2.5-1 for Electrical Related Items
Insulated cables and connections were not specifically addressed in the SBO open item response, but the response did not change the fact that no scoping was performed for cables as stated in Application Sections such as 2.1.1.1.3, 2.1.1.2.3 and 2.1.1.3.5:

> "No scoping was performed for insulated cables and connections and all insulated cables and connections are in scope as part of a bounding scope."

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 cables in these additional areas are included in the aging management review for insulated cables and connections submitted in the June 2001 License Renewal Application. This June 2001 cable aging management review is a bounding review that included all cables installed in these additional areas and structures (the areas and structures now identified as being within scope). The aging management programs identified for insulated cables and connections apply to the cables (that meet the scope of each program) installed in these additional areas and structures.

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

Duke Response to Open Item 3.6.1-1
Open Item 3.6.1-1 asks to justify the adequacy of visual inspections, or to say it another way: How we know that "looking" is good for these specific cables.

The topics we will cover include:
- The basic function of high range radiation monitor and neutron monitoring instrumentation
- Where most high range radiation monitor and neutron monitoring instrumentation circuit failures occur
- High range radiation monitor and neutron monitoring cables that are the subject of SER Open Item 3.6.1-1
- Where the subject high range radiation monitor and neutron monitoring cables are installed
- The strength of visual inspections
- How visual inspections compare with electrical tests for instrumentation cables regarding the detection of aging degradation

Cables to which SER Open Item 3.6.1-1 Applies

The primary function of high range radiation monitor and neutron monitoring circuits is to measure the amount of radiation in an area[1] and provide this information for plant operations.

Most reported failures of high range radiation monitor and neutron monitoring cables and connections have occurred in portions of the instrumentation circuits installed in the Reactor Building and specifically those portions located in close proximity to the reactor pressure vessel.[2]

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1 Table B-1(page B-15) of *License Renewal Electrical Handbook:* EPRI, Palo Alto, CA: 2001. 1003057.

2 Cable AMG Section 3.7.2.3 (page 3-36), Neutron Monitoring Systems, states that the most significant cause of failure was exposure to high temperature and that this is consistent with the characteristically severe thermal and radiation environment, relatively high level of moisture and limited space available in the areas in close proximity to the reactor pressure vessel. The Cable AMG is the *Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations*, SAND96-0344, September 1996.

SER Open Item 3.6.1-1 pertains only to non-EQ cables. All areas within the Reactor Buildings are EQ "harsh" environment areas.[3] Due to the similarity in scope between the License Renewal Rule and the EQ Rule for "harsh" environment areas [4], high range radiation monitor and neutron monitoring cables installed in the Reactor Buildings that are within License Renewal scope are also within EQ scope and are included in the plant EQ program.

Because the AMR applies only to non-EQ components, SER Open Item 3.6.1-1 pertains only to high range radiation monitor and neutron monitoring cables installed in the non-"harsh" areas of the Auxiliary Buildings and other areas outside the Reactor Buildings.

The EQ program maintains the intended functions of in-scope high range radiation monitor and neutron monitoring cables installed in the Reactor Buildings, where most reported failures of high range radiation monitor and neutron monitoring cables and connections have occurred.

Portions of the high range radiation monitor and neutron monitoring instrumentation circuits located in non-"harsh" environment areas of the Auxiliary Buildings and other plant areas are routed in the normal open cable tray system along with other accessible cables included in the visual inspections program.

The visual inspections program is patterned after the inspections done in preparation for the Oconee license renewal application. These inspections were very successful and led to the generation of the EPRI *Adverse Localized Equipment Environment Guideline* that is referenced in the GALL Report. These inspections were successful in identifying aging of cables at all levels of degradation from mild to severely degraded.

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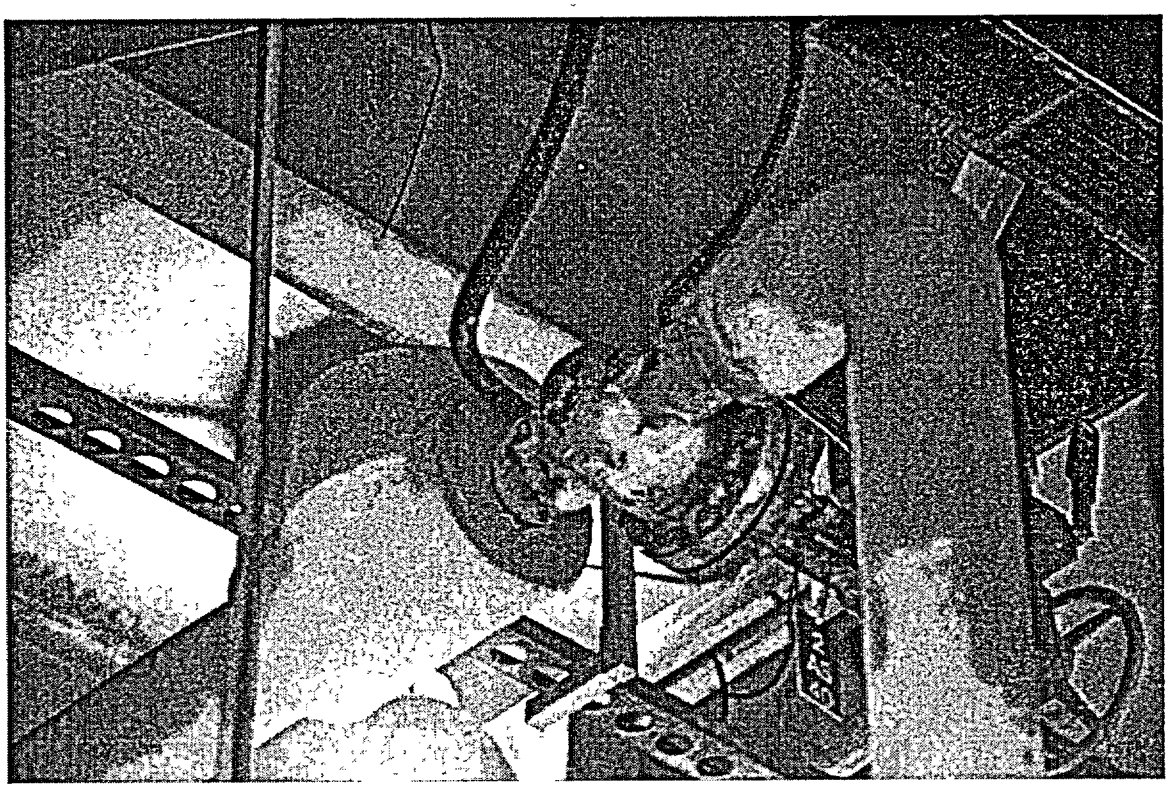
3 Harsh Environment: An environment expected as the result of the postulated service conditions for the station design basis and post-design basis accidents. Harsh environments are the result of a loss of cooling accident (LOCA), high energy line break (HELB) or post-LOCA recirculation radiation. Harsh environment equipment is equipment installed in a harsh environment area that must function to mitigate the consequences of the accident as reported in Duke Power's Response to NUREG-0588. From Section 303.1.5, *Definitions*, of Duke Power Nuclear Policy Manual, Nuclear System Directive 303, *Environmental Qualification Program.*

4 10CFR 54.4 and 10CFR 50.49 define the same scope of components installed in a "harsh" area except for the regulated events. High range radiation monitor and neutron monitoring cables within license renewal scope are those that are credited for accident mitigation and they serve the same functions for any of the LR scoping regulated events as are needed to meet the §54.4 safety-related functions. There is no LR-EQ scope difference regarding high range radiation monitor and neutron monitoring cables installed in EQ "harsh" environment areas.

The Strength of Visual Inspections

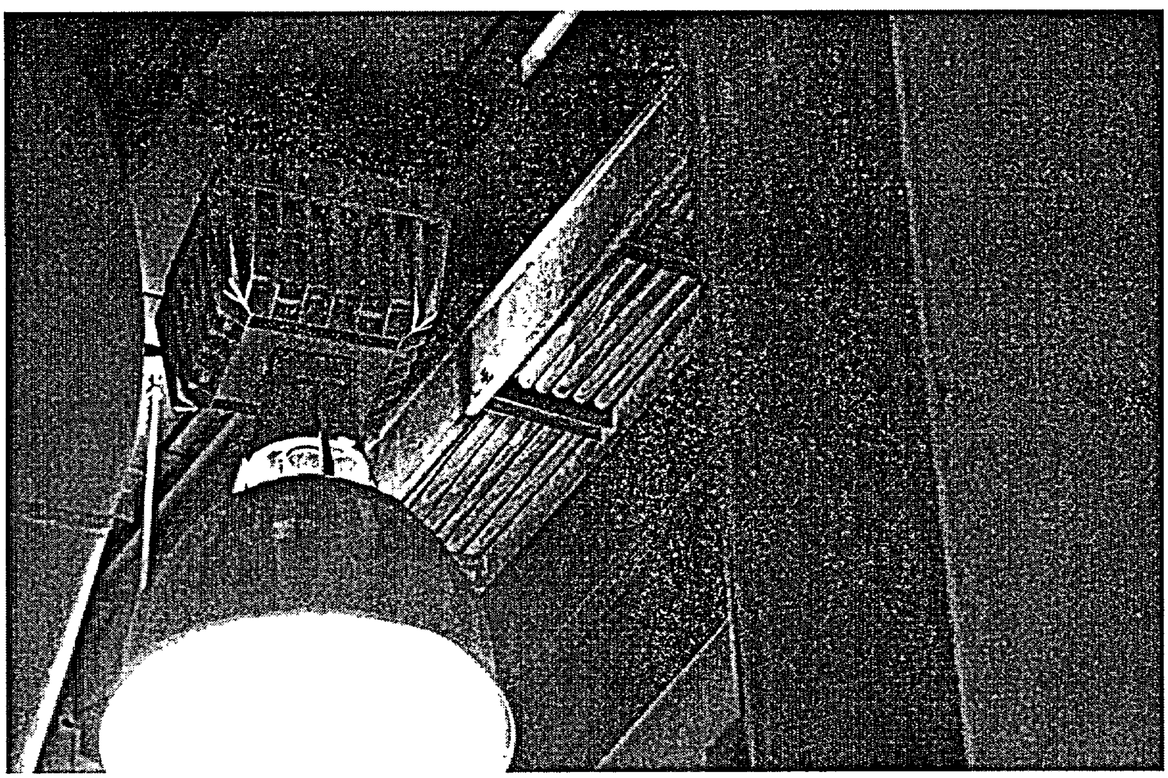
A June 28, 2002, memorandum from the NRC Office of Nuclear Regulatory Research (RES) to the Office of Nuclear Reactor Regulation (NRR), "Technical Assessment of Generic Safety Issue (GSI) 168, *Environmental Qualification of Low-Voltage Instrumentation and Control (I&C) Cables*." Among other things, RES concluded, "*Licensee walkdowns to look for any visible signs of anomalies attributable to cable aging, coupled with the knowledge of operating environments, have proven to be effective and useful.*"

Visual inspections have been proven effective because ***visual inspections find potential problems***. Problems that have not developed to the point of component failure can be identified through visual inspections. Visual inspections observe equipment installation configurations that could cause problems in the future.



The photograph above was taken during the Oconee visual inspection walkdowns that led to the generation of the EPRI *Adverse Localized Equipment Environment Guideline*. This picture shows the power and control cables for an auxiliary steam (AS) system valve laying on top of an uninsulated portion of the pipe. Contact with the hot steam pipe would eventually have degraded the cables, potentially leading to a future failure. The pictured valve is installed near the ceiling in the Turbine Building, some 30 feet above the floor and would only have been found through dedicated visual inspections.

Visual inspections walkdowns also identified the configuration pictured below where a small cable tray with safety-related cables are installed near the ceiling in close proximity to a high-intensity light fixture. What was noted were the concentric ring noted on the bottom of the cables. At the time this was identified it could not be determined if the rings were caused by variations in the light shining on the cables or whether heat from the lamp had discolored the cable jackets.



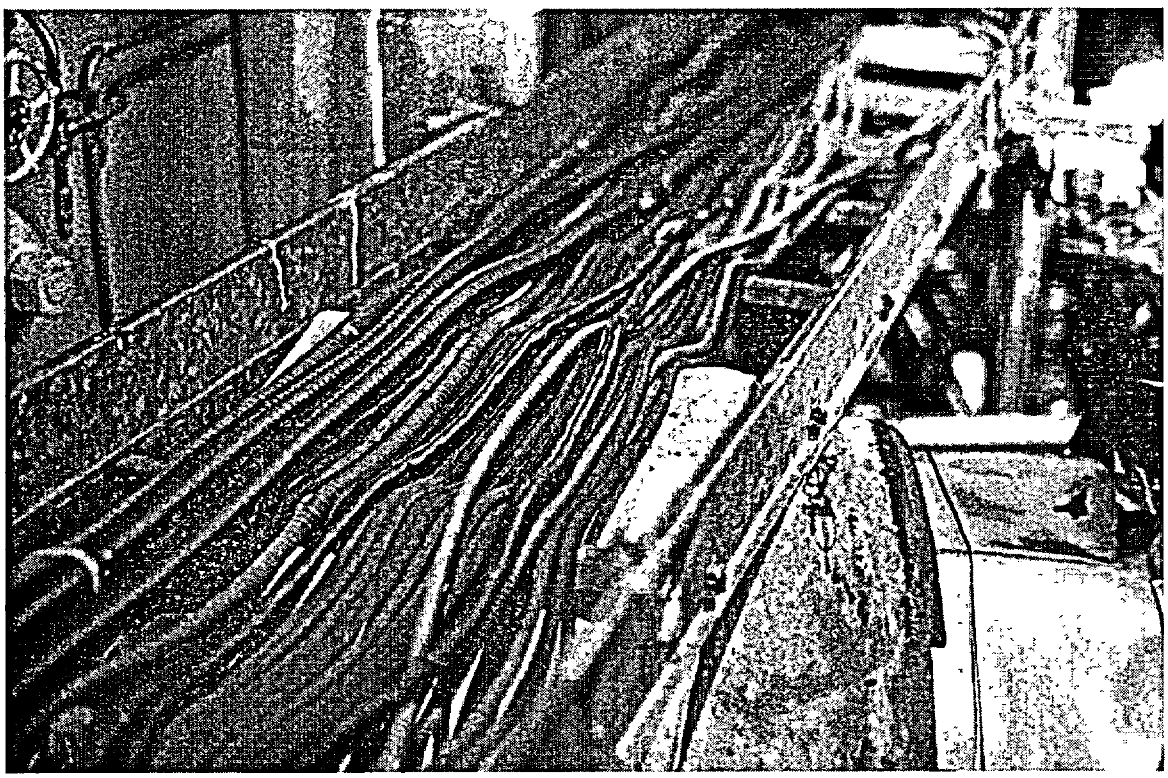
Further investigation revealed that the heat had indeed caused discolorations in the cable jackets, but this was only a surface effect is not a long-term potential problem.

Although this situation was found not to be a problem the configuration of cables in close proximity to high-intensity lamps was noted as something to look for in other areas of the plant and for future visual inspections.

Mechanical Vs. Electrical Property Changes

Section 5.2.2 of the Cable AMG states, "*...mechanical properties must change to the point of embrittlement and cracking before significant electrical changes are observed...*". Embrittlement and cracking are signs of extensive aging that are easily detectable by visual inspection. Signs of less extensive aging, such as discoloration, are also easily detectable by visual inspection. Visual inspections can detect aging degradation early in the aging process before significant aging degradation or failure has occurred.

The photograph below, taken during the Oconee walkdowns, pictures instrumentation cables in a Reactor Building cable tray installed directly over a feedwater line. The heat escaping from the shield wall penetration sleeve around the pipe is accelerating the aging of the cable insulation.



The visual signs that indicate aging degradation of cables in the tray are the way the cables "droop" between the cable tray lattice supports and many of the cable jackets look "dry" and have surface cracks.

As part of the corrective action process, the cables in the tray were tested and all cables are fully functional. If these cables had been only functionally checked, their degraded condition would not have been identified. Without visual inspections, the aging problem with these cables would not be known until one of the circuits failed and the failure was investigated.

Cable AMG Supports Visual Inspections for Neutron Detecting Circuits

As discussed in the SER regarding this open item, the Cable AMG (SAND96-0344) reviews neutron monitoring systems in Chapter 3. The significant and observed aging effects are displayed in Table 4-18. Applicable maintenance, surveillance and condition monitoring techniques are outlined in Table 5-6, which identifies visual inspections for neutron detecting cables and connectors.

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

Duke Response to Confirmatory Item 3.6.1-1
In response to Confirmatory Item 3.6.1-1, the following statement will be added 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 UFSAR Supplement:

Corrective actions should consider the potential for moisture in the area of degradation.

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.

Duke Response to Confirmatory Item 3.6.2-1

In response to Confirmatory Item 3.6.2-1, the summary description of the *Inaccessible Non-EQ Medium Voltage Cables Aging Management Program* provided in Duke response to staff Potential Open Item B.3.19.2-1 (as provided in Duke letter dated July 9, 2002, Attachment 1, pages 89-91) will be inserted in each station's UFSAR supplement in place of the program description previously provided.

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

Duke Response to Confirmatory Item 4.4-1
Duke notes that revisions to NUREG-0933 may not be available in a timely manner to support the license renewal review schedule and that NUREG-0933 does not constitute requirements for licensees to take any specific action. However, in response to Confirmatory Item 4.4-1, Duke proposes the following commitment as an alternative to that proposed by the staff:

> If the staff issues 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 will supplement its license renewal application. 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 safety evaluation report for license renewal January 6, 2003.

The above commitment reflects the appropriate sequence of events for preparing any supplement to the application and is consistent with the notification requirements of §54.21(b), "CLB changes during NRC review of the application." This commitment will be provided in the formal comments to the SER with open items.

Informal Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

* Reactor Coolant System Related Items *

09/17/2002

Enclosure 4

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

Duke Response to New Open Item 3.0.3.10.2-1
In response to New Open Item 3.0.3.10.2-1, Duke agrees with the staff position that volumetric examination of a sample of small-bore Class-1 piping is needed to demonstrate that the effects of aging are being adequately managed. Duke notes that the implementation of a Risk-Informed Inservice Inspection Program that meets the requirements of WCAP 14572 provides for volumetric examination of small-bore Class-1 piping:

- As discussed in Appendix B page B.3.20-5 of the Application, Duke has proposed that aging of small bore piping (piping less than 4-inch NPS) be managed by Risk Informed Inservice Inspection requirements. The risk-informed approach is based on WCAP 14572 Revision 1-NP-A and consists of the following two essential elements: (1) a degradation mechanism evaluation is performed to assess the failure potential of the piping under consideration, and (2) a consequence evaluation is performed to assess the impact on plant risk in the event of a piping failure.

- Duke submitted a request for approval of risk informed methods for McGuire Units 1 and 2 on June 26, 2001 and supplemented information January 11, 2001 and March 15, 2001, as is required by WCAP 14572 Revision 1-NP-A.

- Relief request for socket weld (Relief Request 01-008) submitted with June 26 request.

- WCAP 14572 Revision 1-NP-A and Code case N-577, "Risk-informed Requirements for Class 1, 2, and 3 Piping, Method A" require volumetric examination of socket welds. Duke requested relief to perform VT-2 due to the fact that volumetric examination of socket welds is inconclusive and impractical due to the geometric limitations imposed by a socket weld.

- NRC approval obtained for Risk-Informed ISI and Relief Request June 12, 2002.

- Risk-Informed ISI will cause Duke to perform volumetric examinations of certain risk significant small bore piping.

- Required examination methods are specified in Chapter 4, Inspection Program Requirements, Table 4.1-1 of WCAP 14572, Revision 1-NP-A. Inspection method is based on damage mechanism.

- Risk informed assessment has not been completed for Catawba. Catawba is expected to have similar results and therefore should have a sample of small bore piping that will be volumetrically examined due to future implementation of risk-informed methods.

For the reasons stated above, Duke believes that the staff concern is effectively addressed by the recently approved RI-ISI program for McGuire. A similar RI-ISI program will be implemented at Catawba.

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

Duke Response to New Open Item 3.0.3.10.2-2

New Open Item 3.0.3.10.2-2 contains two specific staff requests. In response to the first request, the following is a list the locations in the McGuire and Catawba reactor coolant system piping that contain welds fabricated from Alloy 82/182 material:

- Pressurizer surge, spray, relief, and safety nozzles weld buildup (Table 3.1-1, page 3.1-9, row 2 of the Application)
- Reactor vessel, primary inlet and outlet nozzles, buttering and welds (Table 3.1-1, page 3.1-11, row 3 of the Application)
- Steam Generator primary nozzle welds (Table 3.1-1, page 3.1-22, row 3, of the Application)
- Auxiliary feedwater nozzle safe end (Alloy 600 Safe End) (Table 3.1-1, page 3.1-25, row 4)

In response to the second request, the following actions have been taken to address the V.C. Summer operating experience as it applies to McGuire and Catawba:

- As part of EPRI MRP Alloy 600 ITG, the Alloy 82/182 Weld Integrity Inspection Committee was formed
- Duke participated in this committee, which recommended that demonstrations be performed to document the capability of automated ultrasonic examination techniques for detecting inside surface-connected flaws in smooth bore nozzle configurations
- The VC Summer hot leg nozzle weld was a field weld (not a machined smooth bore nozzle configuration as is the design at Catawba and McGuire).
- If the weld surface is not smooth good contact cannot be maintained between the UT probe and the weld, which causes inaccurate results

- The geometry of the VC Summer weld was identified as a contributing factor in UT not identifying some of the part depth axial flaws in the hot leg nozzle weld
- Vendors that perform these examinations (Framatome ANP for Duke) performed examinations on a mock-up to demonstrate the effectiveness of their examination techniques. Framatome ANP results were found to be acceptable. The results are documented in EPRI 1006225, "Automated Ultrasonic Inside Surface Examinations of Reactor Coolant System Alloy 82/182 Nozzle Welds Performed in Spring 2001"
- McGuire Unit 1 results of 10 year ISI nozzle to safe weld examinations are documented in EPRI 1006225 (page 4-3)
- McGuire Unit 2 and the Catawba units will have similar inspections during their 10 year ISI

<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 requests that the applicant amend the Pressurizer Spray Head Examination to state that VT-1 examination methods, which are capable of detecting and resolving cracks in the pressurizer spray heads, will be used for the one-time inspection. The scope of this open items includes the potential need to revise the acceptance criteria for this program and the FSAR Supplement summary description.

Duke Response to New Open Item 3.1.2.2.2-1
Duke disagrees with the staff position that a VT-1 inspection is required for the pressurizer spray head for the following reasons:

- The Pressurizer Spray Head is cast austenitic stainless steel
- The aging effect of concern is cracking due to thermal embrittlement
- Small tight cracks will not result in loss of spray function
- The function is spray. To cause the spray head to fail this function would at least require very large cracks and possibly large pieces of the spray head to crack off – conditions readily assessable by the criteria of a VT-3 examination.
- There is no pressure boundary or structural function associated with the spray head – therefore very little stress on the spray head
- The spray head is not a safety related component, and is not relied on to mitigate design basis events. It is credited for cool down and depressurization of the plant from Mode 4 to 5 during a postulated fire event. Cooldown and depressurization could be achieved by other methods if required.
- Duke proposed a VT-3 examination consistent with what was found acceptable for Oconee in NUREG-1723 page 3-115. Even though the spray head is a different design it is also made of CASS and subject to the same aging effects.

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

Duke Response to New Open Item 3.1.3.2.2-1

For McGuire Unit 1:

- Capsule W is a standby capsule and is being used to support a sister plant
- Capsule W has the same weld material as the limiting material of the sister plant
- Capsule W contains material which is not the limiting material for McGuire Unit 1
- Capsule W is not necessary to adhere to 10CFR50 Appendix H or E-185 withdrawal schedule for McGuire Unit 1
- Presently, it is planned to withdraw Capsule W during EOC 18, which will have a little less than 2 times the EOL surface fluence of McGuire Unit 1.

For Catawba Unit 2:

- Capsule U is not necessary.
- The EOL predicted shift (Δ RT_{NDT}) is less than 100° F, therefore only 3 capsules are required to meet the requirements ASTM E-185 referenced in 10 CFR 50, Appendix H.
- Catawba Unit 2 is utilizing 5 capsules for its surveillance program.

For all McGuire and Catawba Units

- As stated in the Application (Section B.3.26), all pulled capsules have either been tested or stored
- Ex-vessel dosimetry program has been established at McGuire and will be installed in the Catawba Units in upcoming outages

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

Duke Response to New Open Item 3.1.3.2.2-2
In response to New Open Item 3.1.3.2.2-2, Duke incorporates by reference (pursuant to §54.17(e)) its response to NRC Bulletin 2002-02 dated September 6, 2002.

The *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* is described in each station's UFSAR Supplement. Each station's UFSAR Supplement will eventually be incorporated into the UFSAR of the respective station following issuance of the renewed operating licenses for McGuire and Catawba.

Any updates/revisions to the *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* as described in each UFSAR that result from the resolution of the issue identified in NRC Bulletin 2002-02 and the Duke responses to this bulletin, will be reflected in the McGuire and Catawba UFSARs in accordance with the Duke program that implements the requirements of §50.71(e). The program summary will then describe the committed aging management activities for not only the initial 40-year license period, but also the period of extended operation.

In addition, the following statements will be added to the description of the *Control Rod Drive Mechanism and Other Vessel Closure Penetration Inspection Program* contained in Section 18.2.6 of the UFSAR Supplement for each station:

The NRC, NEI, MRP, and the nuclear industry are developing actions to monitor for, detect, evaluate, and correct cracking of the Alloy 600 vessel head penetration nozzles of U.S. PWRs. Duke commits to implement the actions developed that specifically relate to McGuire [Catawba] to ensure the integrity of the Alloy 600 vessel head penetration nozzles during the period of extended operation.

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

Duke Response to New Open Item 3.1.4-1(a)
Duke disagrees with the staff position that all of the McGuire and Catawba reactor vessel internals should be inspected for the following reasons:

- All the McGuire and Catawba reactor vessel internals are manufactured by Westinghouse – not by the reactor vessel manufacturers.
- McGuire 1 leads McGuire 2 in operating hours and is clearly the lead plant for reactor vessel internals inspection.
- The Catawba internals will have much less potential for the referenced aging effects, since it is an original upflow design with cooling holes for the baffle bolts and pressure relief holes in the baffle plates. The stresses are also less due to the lower differential pressure across baffle plates from the bypass region.
- The only significant weld in the McGuire and Catawba reactor vessel internals is the circumferential weld in the core barrel, which has a much lower fluence than the regions of concern. All other welds are used to capture locking devices. The core barrel and thermal shield bolts which are in the Oconee internals are not part of the McGuire and Catawba design.
- With all the inspections already committed to for license renewal plants, additional inspections beyond the code inspections should not be necessary for similar internal designs.
- If aging effects are discovered at Oconee or McGuire 1 there will be time to evaluate the potential for other units to experience the same effect and how best to manage the issue.
- Duke response to RAI B.3.27-1 provides a table comparing power level, materials of construction, temperatures and estimated fluences for Oconee Unit 1 and all McGuire and Catawba units.

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

From body of the SER (page 3-142): [Acceptance Criteria] The applicant stated that the acceptance criteria will be based upon analyses and inspections. The applicant stated that the critical crack size for RV internal plates, forgings, and welds, and RV internals made from CASS will be determined by analysis before inspection. For RV internal baffle bolts any detectable cracking on baffle bolts will be unacceptable. The number of baffle bolts needed to be intact and their locations will be determined by analyses. The applicant did not provide any acceptance criteria is provided for the dimensional change effects that could be induced by void swelling. The applicant's acceptance criteria for the RV internals inspection program is incomplete. The staff therefore needs additional information regarding the acceptance criteria for the inspections that are proposed to the RV internals. This issue is characterized as open item 3.1.4-1(b).

Duke Response to New Open Item 3.1.4-1(b)
In response to New Open Item 3.1.4-1(b), the Acceptance Criteria attribute of the Reactor Vessel Internals Inspection summary description contained in each station's UFSAR Supplement will be revised to read as follows (revision text underlined):

> **Acceptance Criteria** – The *Reactor Vessel Internals Inspection* includes the following acceptance criteria:
>
> For the items comprised of plates, forgings, and welds, critical crack size will be determined by analysis <u>and submitted to the NRC staff</u> prior to the inspection.
>
> For baffle bolts, any detectable crack indication is unacceptable for a particular baffle bolt. The number of baffle bolts needed to be intact and their locations will be determined by analysis.

For items fabricated from CASS, critical crack size will be determined by analysis. Acceptance criteria for all aging effects will be developed and submitted to the NRC staff prior to the inspection.

For items subject to dimensional changes due to void swelling, activities are in progress to develop and qualify the inspection method. Acceptance criteria will be developed and submitted to the NRC staff prior to the inspection.

<u>**New Open Item 3.1.4-1(c)**</u> The staff requests that Duke 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.

Duke Response to New Open item 3.1.4-1(c)
In response to New Open item 3.1.4-1(c), the following statement will be added to the plates, forgings, and welds visual inspection portion of **Monitoring & Trending** attribute of the *Reactor Vessel Internals Inspection* summary description contained in each station's UFSAR Supplement:

> 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.

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

Duke Response to New Open Item 3.1.5-1

In response to New Open Item 3.1.5-1, the following changes will be incorporated into the UFSAR Supplements for each station:

(1) In Table 18-1, for the *Steam Generator Surveillance Program*, "18.3" will be added to the entry in the "UFSAR/ITS Location" column.

(2) New Section 18.3 will be added (the References section will become Section 18.4) and the following statement will be included in Section 18.3:

The inspections of the steam generators follow the recommendations of NEI 97-06, "Steam Generator Program Guidelines."

New Open Item 4.2-1 Request for revised PTS value for McGuire Unit 1 due to recently released capsule surveillance data

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

From the body of the SER (page 4-15) Based upon this excerpt from the staff's safety evaluation regarding the reactor vessel nozzles at McGuire, the staff concludes that the applicant need not address this issue for McGuire 1. However, underclad cracking remains a concern for McGuire 2. The applicant is relying upon ultrasonic inspection for resolution of this issue. However, the staff believes that ultrasonic inspection is not effective at detecting defects of the size generated by this phenomenon. Therefore, this issue can be resolved for McGuire 2 only by analysis. For this reason, this issue is characterized as open item 4.3-3 and applies to McGuire 2 only.

Duke Response to Open Item 4.3-3

Duke disagrees with the staff conclusion that the issue of reactor vessel nozzle underclad cracking can only be resolved by analysis for the following reasons:

- Under clad cracking is not a TLAA for either McGuire or Catawba:
    - All of the six criteria of §54.21(c) are not met
    - TLAA is a design-based calculation with parameters limited to 40-years of operation
    - ASME Section XI under clad crack preservice evaluation is not a design, but rather a field-based construction check
    - If this "field-check" had been unacceptable under ASME XI IWB-3500 "field value" threshold criteria, then a design-based calculation (flaw growth) could have been used and then there would be a TLAA for renewal
- The issue of reactor vessel nozzle under clad cracking was identified, addressed, and resolved during the initial licensing of both McGuire units
- Chronology of events and excerpts from relevant correspondence is described in Duke letter dated July 9, 2002
- Copies of the two Westinghouse reports (one for McGuire 2 and one for Catawba 1) referenced in the letter were provided to the staff
- These reports are listed as references in WCAP-15338, "A Review of Cracking

Associated with Weld Deposited Cladding in Operating PWR Plants"

- Each report provides substantial details relative to:
  - Nozzle cladding procedure (Fabrication process)
  - Ultrasonic examination procedure
  - Calibration, raw data sheets, and ultrasonic reflector plots
- The nozzle cladding fabrication process described in each report is essentially the same – post-weld heat treatment / stress relief time is slightly longer for McGuire Unit 2 than Catawba Unit 1.

In conclusion, Duke believes that these reports provide sufficient information to conclude that reactor vessel nozzle under clad cracking is not an issue today – consistent with the staff's conclusion approximately 20-years ago – and is not an aging effect of concern for license renewal and the period of extended operation

Informal Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

* Thermal Fatigue Related Items *

09/18/2002

Enclosure 5

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 is characterized as an open item.

Duke Response to Open Item 4.3-1
No Duke action is required to resolve Open Item 4.3-1.

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

Duke Response to New Open Item 4.3-2

- Duke has described its Thermal Fatigue Management Program (TFMP)
- The TFMP will address the effects of the coolant environment on fatigue life
- Analyses will be performed prior to year 40 as permitted by Part 54 and the SRP-LR
- Response to New Open Item 4.3-4 provides the UFSAR Supplement summary description of the TFMP

- Duke understands that New Open Item 4.3-2 pertains to the following transients listed in Table 4.3-1(C1) and (C2) of the Duke response to RAI 4.3-1 dated 04/15/02:
    - Transient Number 19 – Charging Flow 50% Increase – 24,000 Design Cycles
    - Transient Number 20 – Charging Flow 50% Decrease – 24,000 Design Cycles
    - Transient Number 21 – Letdown Flow 40% Decrease and Return to Normal – 2000 Design Cycles
    - Transient Number 22 – Letdown Flow 60% Increase – 24,000 Design Cycles
- Duke has reviewed the existing engineering calculations for these transients and confirmed that the analyst evaluated these transients as insignificant
- Duke has also reviewed the actual Temperature/Time histories and confirmed that the transients are insignificant
- The design stresses on the charging nozzles due to these transients are approximately zero

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

Duke Response to New Open Item 4.3-4
In response to New Open Item 4.3-4 , the summary description of the *Thermal Fatigue Management Program* in each station's UFSAR Supplement will be completely revised to read as follows:

<u>Introduction Paragraphs (Station specific):</u>

Metal Fatigue UFSAR Supplement
McGuire Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 5.2.1 to assure that components are maintained within design limits. This requirement is managed by the McGuire *Thermal Fatigue Management Program.*

Metal Fatigue UFSAR Supplement
Catawba Technical Specification 5.5.6 establishes the requirement to provide controls to track the number of cyclic and transient occurrences listed in UFSAR Section 3.9.1 to assure that components are maintained within design limits. This requirement is managed by the Catawba *Thermal Fatigue Management Program.*

<u>Summary Descriptions (Generic)</u>

1.0 Thermal Fatigue Management Program
The four key actions of the *Thermal Fatigue Management Program* are:
1.1 <u>Determining the Thermal Cycles to be Monitored and Their Character and Number of Allowed Occurrences:</u> The set of transient events to be managed by the *Thermal Fatigue Management Program* is derived from the associated component information. Included are their thermal and pressure profile characteristics and the minimum of the numbers of occurrences used in the evaluations. As updates occur to associated component information such as analyzed conditions, operational practices, inservice inspection results, or, fatigue

environmental effect modifications required for the extended period of operation (after 40 years), the set of transients and their limits may require revision.

1.2 Monitoring the Thermal Cycles Experienced: From continual monitoring of plant operating conditions, plant conditions that meet the definition of a transient cycle defined by this program are noted. Upon discovery of each transient cycle required to be documented by the program, the cycle count for that transient event is updated. For those events that are logged, the *Thermal Fatigue Management Program* specifies appropriate parameters such as minimum/maximum temperature limits and rates of temperature change that are assumed in the analysis. The logging process captures these values for review.

1.3 Comparison of Observed Events to Allowable Events: For the transients that have occurred since the previous assessment, two evaluations are performed to determine if parameters are within limits. The first evaluation compares the observed values for those parameters applicable to each transient to the limits described in the *Thermal Fatigue Management Program* (e.g. a maximum or minimum temperature limit). The second evaluation is a comparison to the allowable number of occurrences.

1.4 Corrective Action and Confirmation Process: Should the thermal and pressure profile for a specific transient be outside of the parameters defined for that transient set or should an allowable cycle count limit for a transient cycle set be approached or exceeded, this is identified to the appropriate engineering group(s) for resolution. The corrective action program is triggered immediately if profile values are exceeded. Similarly, the corrective action program is triggered if the number of events is expected to exceed the thermal fatigue basis limits within a manageable time period. A manageable time period is the time needed to complete actions to ensure the affected components stay within acceptable cycle count limits.

2.0 Future Modification to the TFMP for Environmentally Assisted Fatigue

The *Thermal Fatigue Management Program* will address the effects of the coolant environment on component fatigue life (environmentally assisted fatigue or EAF) by assessing the impact of the reactor coolant environment on a sample of critical locations selected from NUREG/CR-6260 and other locations expected to have high usage factors when considering environmentally assisted fatigue. The objective to meet in choosing locations will be to ensure by example that no plant location will have an EAF-adjusted CUF that exceeds 1.0 in actual operation.

The sample of critical components can be evaluated by applying the environmental correction factors to the existing ASME Code fatigue analyses and either (1) computing and tracking an

EAF adjusted CUF against an allowable of 1.0 or (2) tracking the instances of transients identified in Paragraph 1.1 above against an EAF adjusted allowable number of transients.

Base formulas for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels. Duke recognizes these formulas as the current methodology for determining such factors.

Since the specific issue of fatigue reactor water effects is only applicable to the period of extended operation for each McGuire and Catawba unit (earliest start date for the extended period of operation for McGuire is 2021 for McGuire 1), Duke anticipates that the equations specified in the subject NUREGs may be superceded to better reflect the then current best practice, including the allowance that these formulas be adjusted by dividing by a Z factor, as defined in EPRI report: *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* (MRP-47). Duke may therefore choose to exercise a different course of action should the NRC approve a less restrictive approach in the future, either through agreement with the industry, or individually with Duke.

The exercise of the above procedure will be at a time prior to the end of the 40th year of each unit's operation. This lead time shall be sufficient to ensure that implementation of corrective actions will prevent the exceedance of 1.0 of EAF-adjusted CUF within the extended period of operation. No requirement exists that any resulting adjustments in allowables be applied prior to the end of the initial 40 years of operation. It is recognized that a discontinuity exists at the 40 year point in the need to apply this adjustment.

References:

- Application to Renew the Operating Licenses of McGuire and Catawba, June 13, 2001 (existing UFSAR Supp Reference 18-1);
- *Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application* (MRP-47)
- M.S. Tuckman (Duke) letter dated April 15, 2002, *Response to Requests for Additional Information in Support of the Staff Review of the Application to Renew the Facility Operating Licenses of McGuire Nuclear Station, Units 1 & 2 and Catawba Nuclear Station, Units 1 & 2*, Docket Nos. 50-369, 50-370, 50-413 and 50-414 (new UFSAR Supp Reference)

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Form 01077 (R3-94)

CERTIFICATION OF ENGINEERING CALCULATION

STATION AND UNIT NUMBER: *Catawba Nuclear Station Unit 1 & 2*

TITLE OF CALCULATION: *Class I Piping Transient Parameter Analysis*

CALCULATION NUMBER CNC-1223.02-00-0001

ORIGINALLY CONSISTING OF:

PAGES 1 THROUGH 250

TOTAL ATTACHMENTS 0 TOTAL MICROFICHE ATTACHMENTS

TOTAL VOLUMES 1 TYPE I CALCULATION/ANALYSIS ☐ Yes ☒ No

TYPE I REVIEW FREQUENCY N/A

THESE ENGINEERING CALCULATIONS COVER QA CONDITION 1 ITEMS. IN ACCORDANCE WITH ESTABLISHED PROCEDURES, THE QUALITY HAS BEEN ASSURED AND I CERTIFY THAT THE ABOVE CALCULATION HAS BEEN ORIGINATED, CHECKED OR APPROVED AS NOTED BELOW:

ORIGINATED BY See Original For these signatures DATE

CHECKED BY DATE

APPROVED BY DATE

ISSUED TO DOCUMENT MANAGEMENT DATE

RECEIVED BY DOCUMENT MANAGEMENT DATE

MICROFICHE ATTACHMENT LIST: ☐ Yes ☒ No SEE FORM 101.4

REV NO	CALCULATION PAGES (VOL) REVISED	DELETED	ADDED	ATTACHMENTS (VOL) REVISED	DELETED	ADDED	VOLUMES DELETED	ADDED	ORIG DATE	CHKD DATE	APPR DATE	ISSUE DATE / RECD DATE
6	2, 174B								[signature] 8/7/96	RCBurg 8/7/96	D.L. Ward 8/11/96	Dtwo 8/11/96 Dg2 5-31-97
7	31, 33, 36, 44, 46, 52, 53, 57, 67	34A	34.1-34.14			ATT 1			RCBurg 3-11-98	[signature] 4-7-98	D. L. Ward 4/7/98	Dtwo 4-7-98

Enclosure 6

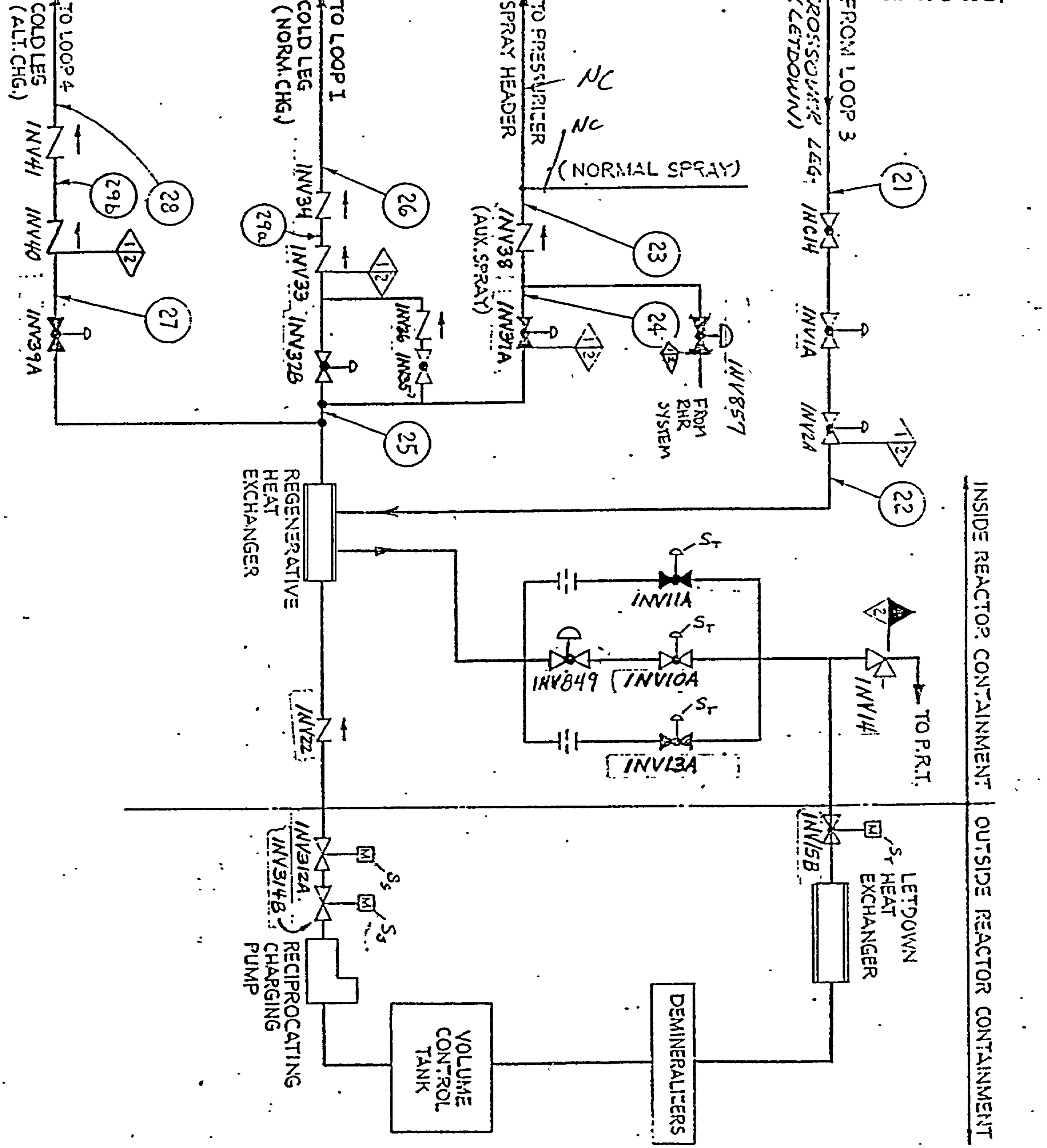
0000 4835 0032

Class I Transient Parameter Analysis

Table of Contents

Chemical and Volume Control System Transient Parameter Analysis
Calculation No. 05-3.0
Catawba R(
Sheet 1 of 5!

Letdown lines and Charging lines



EVENT: CHG FLOW 50% DECREASE (NORMAL CONDITION)

NUMBER OF OCCURRENCES: 24,000 (NOTE 13)

21	P	0 50 175 300 2200 2135 2260 2220
	T	0-1200 560
	F	0-1200 120
22	T	0-1200 560
23	P	SAME AS 26
	T	0-1200 560
	F	0-1200 0
24	P	AMB
	T	AMB
	F	0
25	T	0 6 36 100 1020 1025 1039 1089 1200 500 526 549 557 560 528 512 500 500
26	P	0 50 175 300 2300 2235 2360 2320
	T	SAME AS 25
	F	0 1 1020 1021-1200 100 50 50 100
27	T	SAME AS 25
28	P	SAME AS 26
	T	SAME AS 25
	F	SAME AS 25

ITEM TRANSIENT ANALYSIS	CLIENT [illegible]
CVCS	PROJECT CATAWBA
REV. 0	
	JOB NO. [illegible]
BY RCB DATE 6-23-78 CKD. [illegible] DATE 6-29-78	ITEM NO. SHT. 34 OF 51

EVENT: CHG FLOW 50% INCREASE (NORMAL CONDITION)

NUMBER OF OCCURRENCES: 24,000 (NOTE 12)

21	P	0 60 75 480 1200 2200 2300 2125 1925 2160
	T	0-1200 560
	F	0-1200 120
22	T	0-1200 560
23	P	SAME AS 26
	T	0-1200 560
	F	0-1200 0
24	P	AMB
	T	AMB
	F	0
25	T	0 4 18 72 100 1020 1023 1035 1066 1123 1200 500 442 421 428 405 405 450 476 492 500 500
26	P	0 60 75 480 1200 2300 2400 2225 2025 2260
	T	SAME AS 25
	F	0 1 1020 1021-1200 50 100 100 50
27	T	SAME AS 25
28	P	SAME AS 26
	T	SAME AS 25
	F	SAME AS 25

ITEM TRANSIENT ANALYSIS	CLIENT
CVCS	PROJECT CATAWBA
REV. 0	
	JOB NO.
BY RCB DATE 6-23-78 CKD. JMD DATE 6-29-78	ITEM NO. SHT. 36 OF 51

EVENT: LETDOWN FLOW 40% DECREASE AND RETURN TO NORMAL (NORMAL OCCURRENCE)

NUMBER OF OCCURRENCES: 2000

21	P	0 - 22300 2200
	T	0 - 22300 560
	F	0 1 21600 21601 - 22300 120 72 72 120
22	T	0 - 22300 560
23	P	0 - 22300 2300
	T	0 - 22300 560
	F	0 - 22300 0
24	P	SAME AS 23
	T	SAME AS 23
	F	SAME AS 23
25	T	SAME AS 26
26	P	0 - 22300 2300
	T	0 28 83 200 300 301 348 424 566 21600 (RETURN TO NORMAL) 21900 21901 21962 22300 500 435 293 375 375 443 514 548 560 560 560 525 500 500
	F	0 - 300 301 - 21900 21901 - 22300 (SEE NOTE 15) 100 52 100
27	T	SAME AS 25
28	P	0 - 22300 2300
	T	SAME AS 26
	F	SAME AS 26

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ITEM TRANSIENT ANALYSIS	CLIENT DUKE
CVCS	PROJECT CATAWBA
REV 0	
	JOB NO.
BY RCB DATE 6-23-78 CKD. J.M. DATE 6-29-78	ITEM NO. SHT. 38 OF 51

EVENT: LETDOWN FLOW 60% INCREASE (NORMAL CONDITION)

NUMBER OF OCCURRENCES: 24,000 (NOTE 14)

21	P	0 50 175 300 2200 2135 2260 2220
	T	0-22400 560
	F	0 1 21900 21901 22400 75 120 120 75 75
22	T	0-22500 560
23	P	SAME AS 26
	T	0-22500 560
	F	0-22500 0
24	P	AMB
	T	AMB
	F	0
25	T	0 21 50-300 307 339-21900 21937 21998 22025-22200 22210 22235 22290-22400 500 520 531 469 440 398 378 375 460 482 500
26	P	0 50 175 300 2300 2235 2360 2320
	T	SAME AS 25
	F	0 300 301 22300 22301-22400 (SEE NOTE 15) 55 55 100 100 55
27	T	SAME AS 25
28	P	SAME AS 26
	T	SAME AS 25
	F	SAME AS 25

ITEM TRANSIENT ANALYSIS	CLIENT
CVCS	PROJECT CATAWBA
REV. 0	
	JOB NO.
BY RCB · DATE 6-23-78 · CKD. [illegible] · DATE 6-29-78	ITEM NO. · SHT 40 OF 51

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from Unit 2 Calc (there was no equivalent page(s) for Unit 1 calc)

TRANSIENT EVENT	DESCRIPTION	NO. OF OCCURRENCES
N-1	HEAT UP / START UP	200
N-2	SHUT DOWN / COOL DOWN	200
N-3	STEADY STATE FLUCTUATION	INFINITE.
N-4	NORMAL CHARGING/LETDOWN SHUT OFF AND RETURN TO SERVICE	60
N-5	LETDOWN TRIP W/PROMPT RETURN TO SERVICE.	200
N-6	LETDOWN TRIP W/DELAYED RETURN TO SERVICE.	20
N-7	CHARGING TRIP W/ PROMPT RETURN TO SERVICE.	20
N-8	CHARGING TRIP W/. DELAYED RETURN TO SERVICE.	20
* N-9	CHARGING FLOW 50% DECREASE	24000
* N-10	CHARGING FLOW 50% INCREASE.	24000
* N-11	LET DOWN FLOW 40% DECREASE AND RETURN TO NORMAL	2000
* N-12	LETDOWN FLOW 60% INCREASE	24000.
U-1	REACTOR TRIP FROM FULL POWER.	400.

ITEM				CLIENT	Duke Power Co.
TRANSIENT EVENTS				PROJECT	Catawba Unit II
					CNC-1206.02-70-0071
BY KAS	DATE 7/20/82	CKD CLS	DATE 7/21/82	JOB NO.	0093-~~101~~ 102
BY ~~THN~~	DATE ~~8/1/79~~	CKD ~~[illegible]~~	DATE ~~9/27/79~~	ITEM NO.	11-NC-~~[illegible]~~ 225 SHT 32 of 117.

CHANGE OVER

EDS NUCLEAR

TRANSIENT EVENT	DESCRIPTION	NO. OF OCCURRENCES
U-2	INADVERTENT AUXILIARY SPRAY	10
U-3	LOSS OF POWER (BLACK OUT WITH NATURAL CIRCULATION)	40
U-4	LOSS OF LOAD W/O IMMEDIATE TURBINE OR REACTOR TRIP	80
U-5	LOSS OF FLOW IN ONE LOOP (TEMP. FOR THE LOOP THAT LOST FLOW)	80
U-6	LOSS OF FLOW IN ONE LOOP (AVERAGE MIXED TEMP. IN ALL LOOPS)	80
U-7	REACTOR TRIP WITH COOLDOWN AND INADVERTENT SI ACTUATION	10
U-8	INADVERTENT RCS DEPRESSURIZATION	20
T-1	TURBINE ROLL TEST	10
T-2	HYDROSTATIC TEST	5
T-3	PRIMARY SIDE LEAK TEST	50

ITEM				CLIENT	Duke Power Co.
TRANSIENT EVENTS				PROJECT	Catawba Unit II
					CNC-1206.02-70-0071
BY KAS	DATE 7/20/82	CKD CLS	DATE 7/21/82	JOB NO.	0093-~~101~~ 102
BY ~~TMN~~	DATE ~~8/1/79~~	CKD. ~~[illegible]~~	DATE ~~9/27/79~~	ITEM NO.	CC-NC-~~[illegible]~~ 225 SHT. 33 of 117

CHANGE OVER FROM [illegible] REV

Event	Seg. No.	No. of Cycles	max. Slopes (F^0/sec)	Range (F^0) or $\dot{T}$	MAX Flow (gpm)	Remarks
N-1	ALL	200	0.03	446	100	INSIGNIFICANT.
N-2	ALL	200	-0.03	-446	100	"
N-3	ALL	∞	0.1/-0.1	6/-6	100	"
N-4	26/28	60	406	406	100	RUN. TEMP. DECAY FOR D.S. (1)
"	29a,b	60	346	346	100	RUN TEMP. DECAY FOR D.S.
N-5	ALL	200	53	106	100	RUN
N-6A	ALL	20	36/-490	108/-490	100	RUN (1st half) RESP. OF STAGNANT BRCH FOR U.S.
N-6B	ALL	20	46/-12	460/-211	100	ENVELOPED BY N-4
N-7	ALL	20	-60/+80	-60/80	100	RUN U.S & D.S. (1)
N-8	ALL	20	460/-25	460/175	100	ENVELOPED BY N-4 (1)
* N-9	ALL	24000	-4.33/6.40	-26/32	100	INSIGNIFICANT
* N-10	ALL	24000	14.5/-15	58/-45	100	"
* N-11	ALL	2000	2.32/-68	-65/-68	100	"
* N-12	ALL	24000	9.0/-9	63/-85	100	"
U-1	ALL	400	-.03/.43	-3/13	100	"
U-2	ALL	10	-0-	-0-	100	"
U-3	ALL	40	-.2/.02	-12/14	100	"

TABLE FOR ENVELOPING OF TRANSIENTS

NOTES: 1) THE UPSHOCKS OF 500°F TO 560°F WHEN THE CHARGING TRIPS FOR EVENTS N-4, N-7, N-8 ARE CONSERVATIVELY ENVELOPED TO N-7 (UPSHOCK)

~~CHANGE OVER FROM CC-NC-25 REV. 1~~

REV	BY	DATE	CHECKED	DATE
~~0~~	~~TMN~~	~~4/14/81~~	~~RT~~	~~5/7/81~~
1	KAS	7/20/82	CLS	7/21/82

Duke Power Co./Catawba Unit II

CNC-1206.02-70-0071

eds nuclear

JOB NO 0093-~~100~~ 102

CALC NO CC-NC-~~25~~ 225

Event	Seg. No.	No. of Cycles	MAX Slope (F^0/sec)	Range (F^0) or T	MAX Flow (gpm)	Remarks
U-4	ALL	80	-1.10	-34	100	INSIGNIFICANT.
U-5	ALL	80	·2.93	42	100	"
U-6	ALL	80	-3.0	-12	100	"
U-7	ALL	10	-41.	-41	100	ENVELOPED BY N-7
U-8	ALL	20	-.02	-3	100	INSIGNIFICANT.
T-1	ALL	10	-1.5	-120	80	"
T-2	ALL	5	0	0	55	"
T-3	ALL	50	0	0	55	"

------TABLE FOR ENVELOPING OF TRANSIENTS

NOTES:

~~CHANGE OVER~~
~~FROM CC-NC-15~~ REV. 1

REV	BY	DATE	CHECKED	DATE
~~0~~	~~TMN~~	~~4/14/81~~	~~RT~~	~~5/7/81~~
1	KAS	7/20/82	CLS	7/21/82

Duke Power Co./Catawba Unit II

CNC-1206.02-70-0071

eds nuclear

JOB NO 0093-102

CALC NO CC-NC-225

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0034 4000

Informal Response to McGuire Units 1 & 2 and Catawba Units 1 & 2 Safety Evaluation Report with Open Items

* Structural Related Items *

09/18/2002

Enclosure 7

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

Duke Response to Open Item 2.3-3

In response to Open Item 2.3-3, Duke would like to summarize its previous responses to this staff concern and provide more information in support of using applicable technical specification surveillance requirements in lieu of the Inspection Program for Civil Engineering Structures and Components suggested by the staff.

As stated in our response to RAI 2.3-4, Duke does not define materials such as ventilation area pressure boundary sealants as structures or components. The guidance provided in NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," states that structural sealants are "considered as subcomponents and are not explicitly called out in the scoping and screening procedures." Furthermore, the Commission in the SOC for the Final Part 54 Rule stated:

> "... the Commission has removed the words "portions of" and similar wording from the Statements of Consideration when it could be misinterpreted to mean a subcomponent piece-part demonstration."

Aging management reviews are required for structures and components – not subcomponents. Although ventilation area pressure boundary sealants are not listed as components in the LRA, the function supported by these sealants is to maintain the building pressure boundary enclosure. The pressure boundary function is maintained by McGuire and Catawba technical specifications, limiting conditions of operation and surveillance requirements. The following information identifies the building enclosures where ventilation area pressure boundary sealants provide a pressure boundary function and the technical specifications which address those pressure boundary enclosures:

- The sealants for the Control Room ventilation pressure boundary enclosure are addressed by surveillance testing to demonstrate compliance with *McGuire Technical Specification 3.7.9.4 and Catawba Technical Specification 3.7.10.4.*
- The sealants for the Auxiliary Building ventilation pressure boundary enclosure are addressed by surveillance testing to demonstrate compliance with *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4.*
- The sealants for the Fuel Building ventilation pressure boundary enclosure are addressed

by surveillance testing to demonstrate compliance with *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4.*
- The sealants for the Reactor Building (annulus) ventilation pressure boundary enclosure are addressed by surveillance testing to demonstrate compliance with *McGuire and Catawba Technical Specification 3.6.10.5.*

To manage the aging of ventilation area pressure boundary sealants, Duke is proposing aging management activities that are different than the Inspection Program for Civil Engineering Structures and Components suggested by the staff. The following evaluations are provided for the technical specification surveillance requirements listed above using the ten-attribute presentation discussed in Appendix B of the LRA. These evaluations demonstrate the effectiveness of each of these surveillances. Based on these evaluations, the aging effects of ventilation area pressure boundary sealants will be maintained consistent with the current licensing basis for the period of extended operation.

MCGUIRE TECHNICAL SPECIFICATION 3.7.9.4 AND CATAWBA TECHNICAL SPECIFICATION 3.7.10.4

The *McGuire Technical Specification 3.7.9.4 and Catawba Technical Specification 3.7.10.4* surveillance requirements verify the integrity of the Control Room enclosure and the inleakage rate (or makeup rate) assumed in the dose analysis. The Control Room positive pressure, with respect to potentially contaminated adjacent areas, is periodically tested to verify proper functioning of the Control Room Area Ventilation System (CRAVS). *McGuire Technical Specification 3.7.9.4 and Catawba Technical Specification 3.7.10.4* surveillance requirements are performance monitoring activities.

Scope – The scope of the *McGuire Technical Specification 3.7.9.4 and Catawba Technical Specification 3.7.10.4* is the Control Room ventilation pressure boundary enclosure, including the ventilation area pressure boundary sealants.

Preventive Actions – No actions are taken as a part of this surveillance to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – *McGuire Technical Specification 3.7.9.4 and Catawba Technical Specification 3.7.10.4* measure the positive pressure of the Control Room ventilation pressure boundary enclosure during the periodic performance test. The testing is performed to identify leakage that could indicate cracking or shrinkage of the sealants.

Detection of Aging Effects – In accordance with the information provided in **Monitoring & Trending,** *McGuire Technical Specification 3.7.9.4 and Catawba Technical*

Specification 3.7.10.4 will detect leakage, which would indicate cracking or shrinkage of the ventilation area pressure boundary sealants.

Monitoring & Trending – The method for measuring the positive pressure of the Control Room ventilation pressure boundary enclosure consists of aligning the CRAVS and then measuring the positive pressure within the Control Room ventilation pressure boundary enclosure. The frequency of the performance test is 18 months.

Acceptance Criteria – The acceptance criteria are provided in *McGuire Technical Specification 3.7.9.4 and Catawba Technical Specification 3.7.10.4.* The acceptance criterion is $\geq$ 0.125 inches water gauge for both McGuire Nuclear Station and Catawba Nuclear Station.

Corrective Action & Confirmation Process – Corrective actions for this program are specified in *McGuire Technical Specification 3.7.9.4 and Catawba Technical Specification 3.7.10.4.* Specific corrective actions, including repair or replacement of the ventilation area pressure boundary sealants, are implemented in accordance with the corrective action program.

Administrative Controls – *McGuire Technical Specification 3.7.9.4 and Catawba Technical Specification 3.7.10.4.* surveillances are implemented by written procedure as required by Technical Specification 5.4.1.

Operating Experience – A review of McGuire and Catawba-specific surveillance records did not identify any instances where the ventilation area pressure boundary sealants had been determined to be degraded.

MCGUIRE TECHNICAL SPECIFICATION 3.7.11.4 AND CATAWBA TECHNICAL SPECIFICATION 3.7.12.4

The *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4* surveillance requirements verify the integrity of the ECCS pump room enclosure in the Auxiliary Building. The ability of the ECCS pump room to maintain a negative pressure, with respect to potentially uncontaminated adjacent areas, is periodically tested to verify proper functioning of the Auxiliary Building Filtered Ventilation Exhaust System (ABFVES). *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4* surveillance requirements are performance monitoring activities.

Scope – The scope of the *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4* is the ECCS pump room ventilation pressure boundary enclosure, including the ventilation area pressure boundary sealants.

Preventive Actions – No actions are taken as a part of this surveillance to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4* measure the slightly negative pressure of the ECCS pump room ventilation pressure boundary enclosure during the periodic performance test.

Detection of Aging Effects – In accordance with the information provided in **Monitoring & Trending,** *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4* will detect leakage, which would indicate cracking or shrinkage of the ventilation area pressure boundary sealants.

Monitoring & Trending – The method for measuring the slightly negative pressure of the ECCS pump room ventilation pressure boundary enclosure consists of aligning the ABFVES and then measuring the pressure within the ECCS pump room ventilation pressure boundary enclosure. The frequency of the performance test is 18 months.

Acceptance Criteria – The acceptance criteria are provided in *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4.* The acceptance criterion for McGuire is $\leq$ -0.125 inches water gauge in the ECCS pump room area relative to atmospheric pressure during the post accident mode of operation. The acceptance criterion for Catawba is to maintain the ECCS pump rooms at negative pressure relative to adjacent areas.

Corrective Action & Confirmation Process – Corrective actions for this program are specified in *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4.* Specific corrective actions, including repair or replacement of the ventilation area pressure boundary sealants, are implemented in accordance with the corrective action program.

Administrative Controls – *McGuire Technical Specification 3.7.11.4 and Catawba Technical Specification 3.7.12.4* surveillances are implemented by written procedure as required by Technical Specification 5.4.1.

Operating Experience – A review of McGuire and Catawba-specific surveillance records did not identify any instances where the ventilation area pressure boundary sealants had been determined to be degraded.

MCGUIRE TECHNICAL SPECIFICATION 3.7.12.4 AND CATAWBA TECHNICAL SPECIFICATION 3.7.13.4.

The *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4* surveillance requirements verify the integrity of the fuel handling enclosure. The ability of the fuel handling enclosure to maintain negative pressure with respect to potentially uncontaminated adjacent areas is periodically tested to verify proper function of the Fuel Handling Ventilation Exhaust System (FHVES). *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4* surveillance requirements are performance monitoring activities.

Scope – The scope of *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4* is the fuel handling ventilation pressure boundary enclosure, including the ventilation area pressure boundary sealants.

Preventive Actions – No actions are taken as a part of this surveillance to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4* measure the slight negative pressure in the fuel handling ventilation pressure boundary enclosure during the periodic performance test.

Detection of Aging Effects – In accordance with the information provided in **Monitoring & Trending,** *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4* will detect leakage, which would indicate cracking or shrinkage of the ventilation area pressure boundary sealants.

Monitoring & Trending – The method for measuring the slightly negative pressure of the fuel handling ventilation pressure boundary enclosure consists of aligning the FHVES and then measuring the pressure within the fuel handling ventilation pressure boundary enclosure. The frequency of the performance test is 18 months.

Acceptance Criteria – The acceptance criteria are provided in *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4.* The acceptance criterion for McGuire is verification that the FHVES can maintain an exhaust flow rate > 8000 cfm greater than the supply flow rate. For Catawba, the acceptance criterion is verification that on FHVES can maintain $\leq$ -0.25 inches water gauge with respect to atmospheric pressure during operation at a flow rate $\leq$ 36, 443 cfm.

Corrective Action & Confirmation Process – Corrective actions for this program are specified in *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4.*

Specific corrective actions, including repair or replacement of the ventilation area pressure boundary sealants, are implemented in accordance with the corrective action program.

Administrative Controls – *McGuire Technical Specification 3.7.12.4 and Catawba Technical Specification 3.7.13.4* surveillances are implemented by written procedure as required by Technical Specification 5.4.1.

Operating Experience – A review of McGuire and Catawba-specific surveillance records did not identify any instances where the ventilation area pressure boundary sealants had been determined to be degraded.

MCGUIRE AND CATAWBA TECHNICAL SPECIFICATION 3.6.10.5.

The *McGuire and Catawba Technical Specification 3.6.10.5.* surveillance requirements verify the integrity of the Reactor Building annulus enclosure. The Annulus Ventilation System (AVS) is required to ensure that radioactive materials that leak from the primary containment into the Reactor Building (secondary containment) following a design basis accident are filtered and adsorbed prior to exhausting to the environment. The AVS establishes a negative pressure in the annulus between the Reactor Building and the steel containment vessel. The *McGuire and Catawba Technical Specification 3.6.10.5.* surveillance requirements are performance monitoring activities.

Scope – The scope of the *McGuire and Catawba Technical Specification 3.6.10.5.* surveillance requirement is the annulus between the Reactor Building and the steel containment vessel, including the ventilation area pressure boundary sealants.

Preventive Actions – No actions are taken as a part of this surveillance to prevent aging effects or to mitigate aging degradation.

Parameters Monitored or Inspected – *McGuire and Catawba Technical Specification 3.6.10.5* measure the AVS train flow rate during the periodic performance test.

Detection of Aging Effects – In accordance with the information provided in **Monitoring & Trending,** *McGuire and Catawba Technical Specification 3.6.10.5.* will detect excessive leakage, which would indicate cracking or shrinkage of the ventilation area pressure boundary sealants.

Monitoring & Trending – The method for measuring the flow rate is to align the AVS in its post- accident configuration and then measure the flow rates. Annulus pressure is monitored and recorded during the test. Excessive flow rates or positive pressures may be indicative of leakage which would be evaluated. The frequency of the performance test is 18 months.

Acceptance Criteria – The acceptance criterion for each station is $\leq$-1.20 inches water gauge in the annulus.

Corrective Action & Confirmation Process – Corrective actions for this program are specified in *McGuire and Catawba Technical Specification 3.6.10.5.* Specific corrective actions, including repair or replacement of the ventilation area pressure boundary sealants, are implemented in accordance with the corrective action program.

Administrative Controls – *McGuire and Catawba Technical Specification 3.6.10.5.* surveillances are implemented by written procedure as required by Technical Specification 5.4.1.

Operating Experience – A review of McGuire and Catawba-specific surveillance records did not identify any instances where the ventilation area pressure boundary sealants had been determined to be degraded.

UFSAR Supplement Revisions

As a result of the aging management review summarized above, Table 18-1 of the McGuire UFSAR Supplement will be revised to include the following item:

Topic	Application Location	UFSAR / ITS Location
Ventilation Area Pressure Boundary Sealants	NA	ITS 3.7.9.4 ITS 3.7.11.4 ITS 3.7.12.4 ITS 3.6.10.5

Table 18-1 of the Catawba UFSAR Supplement will be revised to include the following item:

Topic	Application Location	UFSAR / ITS Location
Ventilation Area Pressure Boundary Sealants	NA	ITS 3.7.10.4 ITS 3.7.12.4 ITS 3.7.13.4 ITS 3.6.10.5

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

Duke Response to New Open Item 3.0.3.11.3-1

In response to New Open Item 3.0.3.11.3-1, the summary description of the *Inspection Program for Civil Engineering Structures and Components* in each station's UFSAR Supplement will be revised to include the following statement:

> Examination and assessment of the condition of a structure is performed using guidance provided in codes and standards such as:
> - NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*
> - ACI 349.3, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*

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

Duke Response to New Open Item 3.0.3.18.3-1

In response to New Open Item 3.0.3.18.3-1, the summary description of the *Underwater Inspection of Nuclear Service Water Structures Program*, **Monitoring & Trending** attribute, in each station's UFSAR Supplement will be revised to include the following statement:

> Examination and assessment of the condition of a structure is performed using guidance provided in codes and standards such as:
> - NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*
> - ACI 349.3, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*
> - ACI 201, *Guide for Making a Condition Survey of Concrete in Service*

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

By electronic communication dated August 29, 2002, the staff indicated that as a results of its review of Open Item 2.5-1,

> "that (1) Equipment Pads (concrete) in sheltered or external environments and (2) Reinforced Concrete Beams, Columns, Floor Slabs, Walls (Relay House Floor) (concrete) in sheltered environment are additional examples of existing open item 3.5-1, which is documented in Section 3.5.1.2.1 of the SER with open items. Therefore, the staff is unable to conclude, for these items, that the effects of aging associated will be adequately managed so that there is reasonable assurance that their intended functions will be maintained consistent with the CLB for the period of extended operation, as required by 10 CFR 54.21 (a) (3)."

Duke Response to Open Item 3.5-1
Duke disagrees with the staff conclusion that these structural components require aging management for the period of extended operation for the following reasons:

The aging management review for concrete components at McGuire and Catawba was complete and thorough. In accordance with the guidance provided in NEI 95-10, which was endorsed by the NRC, the aging management review considered both the operating and environmental conditions of the components when determining the aging effects. To further validate the results of the aging management review, operating history was reviewed to identify aging effects that have occurred and could impact the intended function of the structural components. These aging management reviews identified aging effects requiring management for components exposed to certain environments. These reviews did not identify aging effects requiring management for all concrete components.

The staff's position that both the operating and environmental conditions are subject to change throughout the period of extended operation is not valid and is in direct contrast to the statements in the Statement of Considerations (SOC) for the Final Part 54 Rule. Operating experience of more than twenty years is sufficient to identify the range of operating and environmental conditions to which the concrete would be exposed. The SOC for the original license renewal rule supports the use of more than 20 years of operation data as sufficient. As stated in the SOC,

> "the NRC believes that the history of operation over the minimum 20-year period provides a licensee with substantial amounts of information and would disclose any plant-specific concerns with regard to age-related degradation."

In addition, the 1995 License Renewal Rule deleted the term of "ARDUTLR" or age related degradation unique to license renewal. In the SOC for this rule change, the Commission stated:

> The use of the term "age-related degradation unique to license renewal" in the previous license renewal rule caused significant uncertainty and difficulty in implementing the rule. A key problem involved how "unique" aging issues were to be identified and, in particular, how existing licensee activities would be considered in the identification of systems, structures, and components as either subject to or not subject to ARDUTLR. The difficulty in clearly establishing "uniqueness" in connection with the effects of aging is underscored by the fact that aging is a continuing process, the fact that many licensee programs and regulatory activities are already focused on mitigating the effects of aging to ensure safety in the current operating term of the plant, <u>and the fact that no new aging phenomena have been identified as potentially occurring only during the period of extended operation</u>. [*Emphasis added]*

Finally, Appendix A.1 of the LR SRP states that "the determination of applicable aging effects is based on the degradation that has actually occurred and those that potentially could cause structure and component degradation." Supposition of degradation that would result in loss of intended function for the extended period of operation which was not been experienced is hypothetical and has been excluded from the Rule.

In conclusion, Duke has provided a reasonable basis, using NRC rules and NEI guidance, for determining those aging effects that would occur for concrete components that would result in loss of the intended function for the extended period of operation.

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

Duke Response to Open Item 3.5-2
By letter dated June 26, 2002, the NRC staff provided a summary listing of Potential Open Items. In response to this staff letter, Duke submitted a letter dated July 9, 2002 and provided additional information on several of the items, including a response to staff concern – RAI 3.5-1 (Open Item) concerning the lack of groundwater monitoring at either McGuire or Catawba.

By letter dated August 14, 2002, the staff provided its SER with open items and continued to indicate that its concern with a lack of groundwater monitoring was an open item.

Subsequently, on September 3, 2002, the following electronic communication was received from the staff:

> The staff has reviewed the July 9 response to Open Item 3.5-1 and has determined that the response is acceptable. Duke 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 Duke did not postulate a change to the environment due to a chemical release. Duke considers such a scenario to be an abnormal event and cites Appendix A-1 of NUREG-1800 (SRP), which states that aging effects for abnormal events need not be postualted [sic] specifically for license renewal. The staff concurs with the characterization of a potential change to the site environment resulting from a chemical release during the period of extended operation as an abnormal event. As such, the staff agrees that the applicant does not need to commit to a corrective action in the event of a chemical release during the period of extended operation.

Accordingly, no further response from Duke is required for this item and Open Item 3.5-2 should be considered resolved.

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

Duke Response to Open Item 3.5-3
The Duke response to Open Item 3.5-3 is provided in two parts: the first part concerns the ice condenser wear slab and the second part concerns the ice condenser crane wall and accessible portions of the ice condenser structural floor.

With respect to the ice condenser wear slab, Duke has performed an additional review of the design of McGuire and Catawba and determined that the ice condenser wear slab is not within the scope of license renewal because it does not perform a license renewal function. The ice condenser slab is described in each station's UFSAR (Section 6.2.2 for McGuire and Section 6.7.1 for Catawba) as follows:

> The wear slab is a concrete structure whose function is to provide a cooled surface as well as to provide personnel access support for maintenance and/or inspection. The wear slab also serves to contain the floor cooling piping.

Therefore, no further aging management review of the ice condenser wear slab is required for license renewal.

With respect to the accessible portions of the ice condenser crane wall and accessible portions of the ice condenser structural concrete floor, Duke disagrees with the staff conclusion that these structural components require aging management for the period of extended operation for the same reasons that Duke provided in its March 11, 2002 response to RAI 3.5-6 and the response to Open Item 3.5-1 provided above.

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

Duke Response to New Open Item 3.5-4

Duke disagrees with the staff. Duke has reviewed the following technical specifications and selected licensee commitments, including the applicable bases:

McGuire
EPL System - Technical Specification (SR) 3.8.4.3
EPQ System - Selected Licensee Commitment 16.8.3.3
EQD System - Selected Licensee Commitment 16.9.7.12
ETM System - Selected Licensee Commitment 16.9.7.17

Catawba
EPL System - Technical Specification (SR) 3.8.4.4
EPQ System - Technical Specification (SR) 3.8.4.4
EQD System - Selected Licensee Commitment 16.7-9.2
ETM System - Selected Licensee Commitment 16.7-9.4

and believes that no changes to any of these documents are required. These documents contain adequate descriptions of the battery rack inspections. For example, McGuire Technical Specification Surveillance Requirement 3.8.4.4 states:

> Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration that could degrade battery performance.

In addition, Technical Specification 5.4.1 requires written procedures for all technical specification surveillance requirements. Commitments contained with Selected Licensee Commitments are also implemented by written procedures. All of these procedures have adequate descriptions of the battery rack inspections.

Finally, these battery rack inspections and implementing procedures were thoroughly reviewed during the NRC aging management review inspection performed at each station. The results of this inspection are documented in NRC Inspection Report 50-369/02-06, 50-370/02-06, 50-413/02-06 and 50-414/02-06. The reports concludes "The Applicant had provided adequate guidance to ensure aging effects will be appropriately managed."

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

Duke Response to New Open Item 3.5-5

In response to New Open Item 3.5-5, the summary description of the *Crane Inspection Program*, **Monitoring & Trending** attribute, in each station's UFSAR Supplement will be revised to include the following statement:

> Examination and assessment of the condition of a structure is performed using guidance provided in codes and standards such as:
> - ANSI B30.2.0, "Overhead and Gantry Cranes," American National Standard, Section 2-2, *Safety Standards for Cableways, Cranes, Derricks, Hoists, Hooks, Jacks and Slings*, The American Society of Mechanical Engineers, New York.
> - ANSI B30.16, *Overhead Hoists (Underhung)*, The American Society of Mechanical Engineers, New York.
> - 29 CFR Chapter XVII, 1910.179, *Occupational Safety and Health Administration, Overhead and Gantry Cranes*.

Informal Response to
McGuire Units 1 & 2 and Catawba Units 1 & 2
Safety Evaluation Report with Open Items

* Mechanical Related Items *

09/18/2002

Enclosure 8

<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.

Duke Response to Open Items 2.3-1 and 2.3-2
Duke disagrees with the staff for the following reasons:

- 10 CFR 54.21(a)(1) notes that damper and fans without exception are excluded from an aging management review.

- NEI 95-10, *NEI 95-10 (Revision 2) Industry Guideline for Implementing the Requirements of 10 CFR 54 – The License Renewal Rule,* that is endorsed by the staff as an acceptable method for implementing the requirements of 10 CFR 54 notes in Appendix B that dampers and fans are not passive, and therefore, they are not subject to an aging management review.

- NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, notes in Table 2.1-5 that dampers and fans are not passive, and therefore, they are not subject to an aging management review.

- NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, does not contain any entries for dampers and fans.

- The above documents all show that dampers and fans are not subject to an aging management review. In preparing the technical work and the Application, Duke followed the industry and staff guidance documents in effect at the time. As a result, Duke determined that the dampers and fans were within the scope of license renewal but not subject to an aging management review. This is the same position taken during the renewal of the Oconee license that determined acceptable by the Oconee SER presented in NUREG 1723.

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

Duke Response to Open Item 2.3.3.12.2-1
The flexible hose in the Diesel Generator Cooling Water System is replaced on a qualified life. The flexible hose is replaced on a six-year frequency. Therefore, the flexible hose is not subject to an aging management review.

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

Duke Response to Open Item 2.3.3.13.2-1
The flexible hose on the inlet and outlet the of the diesel generator crankcase vacuum blowers are replaced based on condition. The synthetic rubber flexible hose is inspected for cracking and signs of wear on a six-year frequency.

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

Duke Response to Open Item 2.3.3.14.2-1
The flexible hoses in the Diesel Generator Fuel Oil System are replaced on a qualified life. The flexible hoses are replaced on a six-year frequency. Therefore, the flexible hoses are not subject to an aging management review.

<u>Open Item 2.3.3.35.2-1</u> 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.

Duke Response to Open Item 2.3.3.35.2-1
The flexible hoses on the Standby Shutdown Diesel Generator Fuel Oil Sub-system are replaced based upon condition. Every eighteen months, the flexible hoses are inspected for cracking and signs of wear.

<u>**New Open Item 3.0.3.2.3-1**</u> 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.

Duke Response to New Open Item 3.0.3.2.3-1

In response to New Open Item 3.0.3.2.3-1, the summary description of the *Chemistry Control Program* in each station's UFSAR Supplement will be revised to include the following statement:

The *Chemistry Control Program* contains system specific acceptance criteria that are based on the guidance provided in:

- PWR Primary Water Chemistry Guidelines: Revision 4, EPRI TR-105714-V1R4, March 1999.
- PWR Secondary Water Chemistry Guidelines – Revision 5, EPRI TR-102134-R5, 2000.
- Closed Cooling Water Chemistry Guideline, EPRI TR-107396, October 1997.
- Catawba Nuclear Station Units 1 and 2, <u>Technical Specifications</u>.
- McGuire Nuclear Station Units 1 and 2, <u>Technical Specifications</u>.
- Catawba Nuclear Station Units 1 and 2, <u>Updated Final Safety Analysis Report</u>, Chapter 16, Selected Licensee Commitments, Sections 16.5-3, 16.7-9, and 16.8-5
- McGuire Nuclear Station Units 1 and 2, <u>Updated Final Safety Analysis Report</u>, Chapter 16, Selected Licensee Commitments, Sections 16.5-7, 16.8-3, and 16.9-7
- Vendor recommendations for water and fuel oil quality

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

Duke Response to New Open Item 3.0.3.9.1.2(a-g)
New open item 3.0.3.9.1.2(a-g) applies to the following heat exchangers:

Open Item Number	Component	SER Section Number	LRA Section Number
3.0.3.9.1.2(a)	Pump Motor Air Handling Units (McGuire only)	3.0.3.9.1.2	B.3.17.6
3.0.3.9.1.2(b)	Pump Oil Coolers (McGuire only)	3.0.3.9.2.2	B.3.17.7
3 0.3.9.1.2(c)	Containment Spray Heat Exchangers	3.2.4.2.2	B.3.17.2.2
3.0.3.9.1.2(d)	Component Cooling Heat Exchangers	3.3.5.2.2	B.3.17.1.2
3.0.3.9.1.2(e)	Control Area Chilled Water Chillers	3.3.8.2.2	B.3.17.4
3.0.3.9.1.2(f)	Diesel Generator Engine Cooling Water Heat Exchangers	3.3.12.2.2	B.3.17.3.2
3.1.3.9.1.2(g)	Diesel Generator Engine Starting Air Aftercoolers (Catawba only)	3.3.17.2.2	B.3.17.5

The acceptance criteria for each Heat Exchanger Preventive Maintenance Program are no unacceptable loss of material that could result in a loss of the component intended function as determined by engineering evaluation. Duke agrees that additional information describing the engineering evaluation that will be used to define "unacceptable loss of material" is needed for the staff to make a reasonable assurance finding with respect to acceptance criteria of the programs. The following details of each Heat Exchanger Preventive Maintenance Program are provided to assist in the finding.

For New Open Item 3.1.3.9.1.2(a), the program credited for managing loss of material for the pump motor air handling units is a new program to be implemented following the issuance of the renewed operating license for McGuire and by June 12, 2021. Because these heat exchanger tubes are a coil design, they are not candidates for eddy current testing. As described in Section B.3.17.6 of the LRA, either destructive or nondestructive examination will be performed that allows examination of the internal surfaces of the tubes. If evidence of loss of material is observed during the initial inspection, a problem report will be developed in accordance with Problem Investigation Process of 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. Duke believes it is premature to specify actual criteria for evaluating severity and the need for corrective actions for a new inspection for which the analysis method is not yet known.

For New Open Items 3.1.3.9.1.2(b) through 3.1.3.9.1.2(g), eddy current testing is the method used to manage loss of material of the heat exchanger tubes. The information that follows describes the acceptance criteria that apply to the existing programs which are the subject of 3.1.3.9.1.2(c) through (g) and will apply to the new program that is the subject of New Open Item 3.1.3.9.1.2(b).

Eddy current testing is an acceptable industry practice used for detecting wall loss in heat exchangers, but requires careful engineering evaluation of all test results to provide the proper management of a heat exchanger. Steam Generators are the only plant heat exchangers for which there exists station Technical Specifications or set of standards that regulate the depth of flaw at which a tube is plugged and removed from service. For the low pressure, low temperature heat exchangers that are the subject of these open items, evaluating eddy current test results for "unacceptable loss of material" involves many variables such as tube material, characterization of the indication in terms of percent wall loss, rate of degradation as compared to previous indications and the frequency of subsequent testing. A greater wall loss range may be considered acceptable for an indication that is tested frequently or that shows little or no degradation from previous tests; a lesser wall loss range may be considered unacceptable if the indication shows significant degradation from previous tests or that is not tested as frequently.

Eddy current testing at Duke is performed by a vendor who specializes in the practice. The vendor supplies an eddy current test report to Duke for each test they perform. The four-step acceptance criteria process described below is used to generate the final test report.

(1) At the conclusion of testing of a component, the vendor's eddy current testing manager reviews the data and makes a plugging recommendation in the preliminary report based on his

assessment of the damage flaws and experience with testing the component. Experience demonstrates that these specialists generally recommend evaluation at around a 70% wall loss range.

(2) Duke then reviews the entire test data provided in the preliminary test report, including the recommendation for plugging, prior to returning the component to service. Duke evaluates the recommendations using all the information they have available. Particularly, Duke evaluates the rate of degradation based on the history of the tube. The wall loss may be deemed acceptable if the tube is showing minimal to no degradation from previous inspections. Consideration is also given to the frequency of the next inspection; if frequent inspection is performed, then a higher wall loss range may be acceptable and if less frequent inspection is performed then lower wall loss range may be unacceptable.

(3) Depending on the type of tubing material and tubing damage detected with eddy current testing and possibly verified with actual tube pulled samples, a wall loss correlation may be determined as a threshold for evaluating the tube for plugging repair. Past operating experience with the type of tubing flaw may also be a very useful factor in determining the wall loss plugging threshold.

(4) The loss of material experienced by these heat exchanger tubes generally manifests itself as pits. These pitting flaws are not very likely to fail heat exchanger tubing due to mechanical stress of pressure and temperature due to the shouldered nature or material reinforcement around pits. Therefore, the pitting rate as determined from past eddy current testing experience becomes the primary factor to consider when selecting tubes to remove from service to prevent later on-line tube leaks.

Duke's experience in evaluating eddy current testing results has proven effective during the operation of McGuire and Catawba. Corrective actions such as tube plugging and even tube bundle and heat exchanger replacement have been taken as a result of failed acceptance criteria of these programs. Duke's experience of using the four-step process of evaluating "no unacceptable loss of material" described above provides reasonable assurance that the aging effects of these heat exchanger subcomponents will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

Duke Identified Mechanical Item 09/18/2002

In the process of reviewing the SER, Duke identified a statement in Section B.3.17.7, *Heat Exchanger Preventive Maintenance Activities- Pump Oil Coolers* that does not clearly reflect Duke's intention with respect to these coolers. The *Heat Exchanger Preventive Maintenance Activities- Pump Oil Coolers* is a new program for license renewal and applies to eight (8) coolers per unit for a total of sixteen (16) coolers at McGuire only. The program description in the LRA states "Non-destructive (NDT) will be performed on 100% of the tubes," which could be misinterpreted. Duke's intention is to perform non-destructive testing on 100% of the tubes of one of the sixteen coolers within the scope of the program.

The selection of the specific cooler to be examined will take into consideration the normal operating environments of the coolers. The reciprocating charging pump and safety injection pump do not run during normal operation and therefore the reciprocating charging pump bearing oil coolers and speed reducing oil coolers and the safety injection pump bearing oil coolers are normally isolated. The centrifugal charging pumps are normally in service and therefore the centrifugal charging pump bearing oil coolers and speed reducing oil coolers should experience the most susceptible service environment for loss of material to occur. One of the centrifugal charging pump's coolers will therefore be examined as a representative of the total scope.

A sample inspection of one of the sixteen coolers is considered acceptable because of the excellent operating experience of these coolers. As described under Operating Experience in Section B.3.17.7 of the LRA, there have been no tube failures in any of the heat exchangers within the scope of this program, as confirmed through periodic leak detection. This leak detection is performed via periodic oil sampling. The sample chosen is appropriate as a leading indicator of other components in the program because it is most likely to experience aging effects. Prior experience in leak detection provides a basis for concluding that the program will be an effective method of monitoring the components during the period of extended operation. Therefore, past operating experience can be relied on to provide the basis for this new program.

Implementing a sample inspection as part of the *Heat Exchanger Preventive Maintenance Activities- Pump Oil Coolers* more closely aligns this program with the *Heat Exchanger Preventive Maintenance Activities- Pump Motor Air Handling Units* described in Section B.3.17.6 of the LRA and found acceptable by the staff in Section 3.0.3.9.1.2 of the SER.

A review of the McGuire UFSAR Supplement revealed that a more clear description should be included for *Pump Oil Coolers* and *Pump Motor Air Handling Units*. The McGuire UFSAR Supplement summary description of the *Pump Oil Coolers* will be revised to add the following statement:

A non-destructive examination will be performed on 100% of the tubes of one of the sixteen coolers within the scope of the program following issuance of renewed licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

In addition, the McGuire UFSAR Supplement summary description of the *Pump Motor Air Handling Units* will be revised to add the following statement:

A destructive or non-destructive examination will be performed on one of the twelve cooling units within the scope of the program following issuance of renewed licenses for McGuire Nuclear Station and by June 12, 2021 (the end of the initial license of McGuire Unit 1).

<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 believes that the applicant should propose an activity to verify that the external surfaces of buried components are not degrading based upon some sampling assessment of most vulnerable locations.

Duke Response to New Open Item 3.0.3.13.2-1
Duke disagrees with the staff for the following reasons:

- Duke disagrees that a sampling assessment of most vulnerable locations will be effective in managing the random nature of externally-generated pitting caused by degradation in the coatings.

- As noted in the program description in the draft SER, buried components in several systems falling within the scope of license renewal are coated and wrapped to protect the external surfaces. Should the coating not be effective in protecting the piping , the external surfaces could age.

- The purpose of this program is to indirectly monitor the condition of that coating in protecting the external surfaces. Duke is not aware of an aging mechanism that will result in coating degradation. With the protective coating in place, no aging effects will occur on the external surfaces of the piping that will cause loss of intended function.

- The existing operating experience indicates that coating problems that have occurred and that have, in turn, lead to pitting in the buried piping are caused by mechanical damage in locations which could not be predicted.

- As demonstrated in the Operating Experience attribute of Section B.3.24.1 of Appendix B of the Application that was repeated in the draft SER, the program is capable of detecting through-wall pits that have been random in nature and have not failed the system intended function. To date, leakage has been attributable to construction defects in the coating that resulted in degradation of the external pipe surface and a water hammer event which cracked a weld.

- Coatings damage may lead to localized corrosion due to the establishment of a galvanic cell at the site of the coatings damage. This localized corrosion can result in a through-wall pit in the pipe. From Duke's operating experience, a visual inspection performed of the external coating after the discovery of a externally generated pinhole lead found the surrounding coating and piping material to be in an acceptable condition.

- No trends exist in the operating experience to suggest visual inspection of the external coating is warranted without specific evidence. The program as defined will trigger such a visual inspection should conditions arise that warrant it. Specifically, an increase in the number and frequency of externally generated through-wall pinholes that would prompt Duke to question the condition of the external coating and would trigger external visual inspections.

- Duke believes that an inspection of approximately 80% of the buried piping surface area for symptomatic evidence of external surface degradation provides a much more robust aging management program longer term than a sampling assessment.

- Further, Duke is not aware of any criteria for developing a sampling assessment that would identify the most vulnerable locations of several thousand feet of buried components that would be truly representative of all locations.

- Finally, The *Preventive Maintenance Activities – Condenser Circulating Water System Internal Coating Inspection* is an equivalent activity that was found acceptable to the staff in NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station,* Units 1, 2, and 3.

<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 intended function, since sufficient flow at prescribed pressures can still be provided by the system. The applicant also state that these localized concerns will lead to structural integrity concerns only when a significant number of pinholes are present and that a trend of indications of through-wall leaks will trigger corrective actions. However, the staff believes that localized corrosion can result in the loss of pressure boundary intended function under a design basis event before the corrosion reveals itself as pinhole leaks. Therefore, the applicant should justify how its program will manage the effects of localized corrosion from pitting and MIC to ensure that the intended pressure boundary function can be maintained under all design basis events consistent with the CLB during the period of extended operation, as required by 10 CFR 54.21(3).

Duke Response to New Open Item 3.0.3.15.2-1
Duke understands that the staff's concern in this new open item is structural integrity of piping systems due to loss of material, in particular localized corrosion, under all design basis conditions.

The Service Water Piping Corrosion Program, formalized as a part of Duke's response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," utilizes ultrasonic technology to look for loss of material which includes both general and localized corrosion. The program includes inspection locations representative of every pipe size, in each analysis model pipe run, for each flow regime, and upstream/downstream of each major piece of equipment. The periodic ultrasonic testing (UT) at these locations will identify any potential areas of severe degradation, including general and localized corrosion, which could exceed the ability of the piping to maintain its structural integrity in a design basis event.

As Duke has previously described, the primary issue addressed by the program is gross wall loss. Gross wall loss is deterioration of material condition sufficiently extensive to lead to structural instability and loss of component intended function. The secondary issue addressed by the program is the gathering of other symptomatic evidence that will serve as anomalous indications of material degradation. An example of such evidence is pinhole leaks caused by pitting and localized corrosion. As made clear by the Code design rules, pitting absent general corrosion is not a structural concern under normal operation or design basis conditions unless there exists a large number of pits in one area. A large number of pits in one area is essentially gross wall loss.

When an occurrence of localized corrosion is identified either through a low UT reading or a pinhole leak, an evaluation is performed to justify its structural integrity under all design basis conditions (in accordance with the appropriate design code and under guidance of NRC Generic Letter 90-05, "Guidance for Performing Temporary Non-code Repair of ASME Code Class 1, 2 and 3 Piping.") Additionally, UT will detect if there are numerous occurrences of localized corrosion in a given sample area because it does "look" like gross wall loss. As described in the Application, occurrences of localized corrosion are trended to assure an awareness of the progression of the material condition.

The staff concern that localized corrosion can lead to a structural integrity concern before it is revealed as pinhole leaks is valid. The Service Water Corrosion Program has been designed to address this concern by performing appropriate inspections, evaluations and trending and by taking appropriate corrective actions. The Service Water Corrosion Program is subject to ongoing regulatory oversight including the Service Water System Operational Performance Inspection (SWSOPI) and the Safety Systems Engineering Inspection (SSEI) McGuire and Catawba both have been inspected in recent years. This aging management program is consistent with the GALL Report and with the similar program at Oconee which the staff has found adequate for license renewal (Reference NUREG-1723, Section 3.2.13). As such the Service Water Corrosion Program can adequately manage loss of material from both general and localized corrosion for the license renewal systems that credit this program so that the component intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

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

Duke Response to New Open Item 3.3.6.2.1-1
The expansion joints are woven polyester and/or nylon fabric coated with chlorobutyl rubber. The expansion joints are located in the yard near the bottom of open pits adjacent to the Turbine Buildings. As a result of being located near the bottom of the open pits, the expansion joints are exposed to limited amounts of ultraviolet radiation. Butyl rubbers, which included chlorobutyl rubber, are essentially immune to damage from ultraviolet radiation and have a continuous temperature rating of 150°F [Reference]. Since the expansion joints are essentially immune to damage from ultraviolet radiation and receive minimal exposure and local temperatures are well below the 150°F, no aging effects requiring management during the period of extended operation were identified.

Reference

Engineering Materials Handbook, McGraw-Hill, Inc., New York, New York, 1958.

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

Duke Response to New Open Item 3.3.17.2.1-1

The response to RAI 2.3.3.17-2 provided by Duke is in error. The carbon steel emergency diesel generator starting air distributor filter is subject to loss of material in a sheltered environment. The table entry provided in the response to RAI 2.3.3.17-2 should be replaced with the following:

Component Type	Component Function	Material	Internal Environment / External Environment	Aging Effect	Aging Management Programs and Activities
Starting Air Distributor Filter	PB	CS	Air (Dry)	None Identified	None Required
			Sheltered	Loss of Material	Inspection Program for Civil Engineering Structures and Components

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

Duke Response to Open Item 3.3.35.2-1
Duke will identify 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 that is managed by the *Chemistry Control Program.* Fouling due to silting is the result of corrosion products being generated throughout the system and deposited in the heat exchanger. The Standby Shutdown Diesel Cooling Water and Jacket Water Heating Subsystems are closed cooling water systems treated with corrosion inhibitors. The corrosion inhibitors preclude the formation of corrosion products. The corrosion inhibitor concentration in the system is monitored by the Chemistry Control Program. The Chemistry Control Program manages fouling due to silting during the period of extended operation by monitoring and maintaining the corrosion inhibitor concentration to preclude the formation of corrosion products.

The second entry in Table 3.3-44, Aging Management Review Results – Standby Shutdown Diesel, on page 3.3-247 of the Application should be replaced with the following:

Component Type	Component Function	Material	Internal Environment External Environment	Aging Effects	Aging Management Programs and Activities
Heat Exchanger Engine Radiator (tubes)	PB, HT	Cu	Treated Water	Loss of Material Fouling	Chemistry Control Program
			Ventilation	None Identified	None Required

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

Duke Response to New Open Item 3.4.1.2.2-1
Section B.3.6 of Appendix B of the LRA provides a description of the Chemistry Control Program. The **Operating Experience** attribute on page B.3.6-4 provides the Duke specific experience to demonstrate the effectiveness of the *Chemistry Control Program* for managing aging effects. A search of the Problem Investigation Process database was performed to demonstrate the effectiveness of the *Chemistry Control Program.* Reports are entered into the database for component failures, relevant industry operating experience, and problems discovered during routine maintenance and testing. This review of operating experience did not reveal any instances of a loss of the component intended functions of the Auxiliary Feedwater System components that could be attributed to the inadequacy of the *Chemistry Control Program.* Additionally, routine maintenance of other secondary side components such as the steam generators and main turbine provide additional operating experience because, although the Auxiliary Feedwater System is normally in standby, it does operate during startup and shutdown and is of the same chemistry as the Feedwater system and other secondary side systems. This good operating experience demonstrates the effectiveness of the *Chemistry Control Program* and does not warrant a one-time inspection.

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

Duke Response to Confirmatory Item 2.3.3.26.2-1
In response to Confirmatory Item 2.3.3.26.2-1, Duke formally provides the following which had been originally by electronic communication on July 16, 2002:

> Catawba technical specification SR 3.7.4.1 applies to the main steam line PORV nitrogen bottles. This technical specification requires that once every 24 hours at least one of the nitrogen bottles on each SG PORV is verified to be pressurized ≥ 2100 psig. This surveillance requirement is performed by a Catawba procedure entitled "Procedure for Checking and Replacing Steam Generator PORV Nitrogen Cylinders and Setting Cylinder Regulators." There are two nitrogen cylinders per SG PORV. Initial pressure in the cylinder is ≥ 2500 psig. This procedure requires that if the pressure in either nitrogen cylinder is less than or equal to 2420 psig, then the nitrogen cylinder is replaced. Replacement cylinders are obtained from a warehouse. The used cylinders are returned to the warehouse. The cylinders are not permanently installed in the plant.