January 28, 2002

Mr. M. S. Tuckman
Executive Vice-President
Nuclear Generation
Duke Energy Corporation
PO Box 1006
Charlotte, NC 28201-1006

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, AND CATAWBA NUCLEAR STATION, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION (LRA)

Dear Mr. Tuckman:

By letter dated June 13, 2001, Duke Energy Corporation (Duke) submitted for Nuclear Regulatory Commission (NRC) review an application, pursuant to 10 CFR Part 54, to renew the operating licenses for the McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2. The NRC staff is reviewing the information contained in this license renewal application and has identified, in the enclosure, areas where additional information is needed to complete its review. Specifically, the enclosed request for additional information (RAI) is from the following section(s) of the LRA:

Section 2.3.1, System Scoping and Screening Results: Reactor Coolant System
Section 3.1, Aging management of Reactor Vessel, Internals, Reactor Coolant System
Section 4.2, Reactor vessel Neutron Embrittlement
Section 4.3, Metal Fatigue
Section 4.7.1, Reactor coolant Pump Flywheel Fatigue
Appendix B, Aging management Programs (Reactor Coolant System)

Please provide a schedule by letter, or electronic mail for the submittal of your response within 30 days of the receipt of this letter. Additionally, the staff would be willing to meet with Duke prior to the submittal of the response to provide clarification of the staff's request for additional information.

Sincerely,

/***RA***/

Rani L. Franovich, Project Manager
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-369, 50-370, 50-413 and 50-414

Enclosures: As stated

cc w/encl: See next page
Mr. M. S. Tuckman
Executive Vice-President

Nuclear Generation
Duke Energy Corporation
PO Box 1006
Charlotte, NC 28201-1006

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE MCGUIRE NUCLEAR STATION, UNITS 1 AND 2, AND CATAWBA NUCLEAR STATION, UNITS 1 AND 2, LICENSE RENEWAL APPLICATION

Dear Mr. Tuckman:

By letter dated June 13, 2001, Duke Energy Corporation (Duke) submitted for Nuclear Regulatory Commission (NRC) review an application, pursuant to 10 CFR Part 54, to renew the operating licenses for the McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2. The NRC staff is reviewing the information contained in this license renewal application and has identified, in the enclosure, areas where additional information is needed to complete its review. Specifically, the enclosed request for additional information (RAI) is from the following section(s) of the LRA:

Section 2.3.1, System Scoping and Screening Results: Reactor Coolant System
Section 3.1, Aging management of Reactor Vessel, Internals, Reactor Coolant System
Section 4.2, Reactor vessel Neutron Embrittlement
Section 4.3, Metal Fatigue
Section 4.7.1, Reactor coolant Pump Flywheel Fatigue
Appendix B, Aging management Programs (Reactor Coolant System)

Please provide a schedule by letter, or electronic mail for the submittal of your response within 30 days of the receipt of this letter. Additionally, the staff would be willing to meet with Duke prior to the submittal of the response to provide clarification of the staff's request for additional information.

Sincerely,

/***RA***/

Rani L. Franovich, Project Manager
License Renewal and Environmental Impacts Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket Nos. 50-369, 50-370, 50-413 and 50-414
Enclosures: As stated
cc w/encl: See next page
DISTRIBUTION:
See next page
Document Name: C:\Program Files\Adobe\Acrobat 4.0\PDF Output\Duke RAI - Reactor Coolant ~.wpd

OFFICE	PM:RLEP:DRIP	LA:DRIP	SC:RLEP:DRIP	BC:RLEP:DRIP
NAME	RFranovich	EGHylton	PTKuo	CIGrimes
DATE	01/24/2002	01/23 /2002	01/28/2002	01/28/2002

OFFICIAL RECORD COPY

DISTRIBUTION:
HARD COPY
RLEP RF
E. Hylton
E-MAIL:
PUBLIC
J. Johnson
W. Borchardt
D. Matthews
F. Gillespie
C. Grimes
J. Strosnider (RidsNrrDe)
E. Imbro
G. Bagchi
K. Manoly
W. Bateman
J. Calvo
C. Holden
P. Shemanski
S. Rosenberg
G. Holahan
S. Black
B. Boger
D. Thatcher
G. Galletti
B. Thomas
R. Architzel
J. Moore
R. Weisman
M. Mayfield
A. Murphy
W. McDowell
S. Droggitis
N. Dudley
RLEP Staff

R. Martin
C. Patel
C. Julian (RII)
R. Haag (RII)
A. Fernandez (OGC)
J. Wilson
M. Khanna
C. Munson
R. Elliott
J. Medoff
M. Razzaque
J. Fair
J. Rajan
S. Hou

McGuire & Catawba Nuclear Stations, Units 1 and 2

Mr. Gary Gilbert
Regulatory Compliance Manager
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Ms. Lisa F. Vaughn
Duke Energy Corporation
422 South Church Street
Charlotte, North Carolina 28201-1006

Anne Cottington, Esquire
Winston and Strawn
1400 L Street, NW
Washington, DC 20005

North Carolina Municipal Power
Agency Number 1
1427 Meadowwood Boulevard
P. O. Box 29513
Raleigh, North Carolina 27626

County Manager of York County
York County Courthouse
York, South Carolina 29745

Piedmont Municipal Power Agency
121 Village Drive
Greer, South Carolina 29651

Ms. Karen E. Long
Assistant Attorney General
North Carolina Department of Justice
P. O. Box 629
Raleigh, North Carolina 27602

Ms. Elaine Wathen, Lead REP Planner
Division of Emergency Management
116 West Jones Street
Raleigh, North Carolina 27603-1335

Mr. Robert L. Gill, Jr.
Duke Energy Corporation
Mail Stop EC-12R
P. O. Box 1006
Charlotte, North Carolina 28201-1006

Mr. Alan Nelson
Nuclear Energy Institute
1776 I Street, N.W., Suite 400
Washington, DC 20006-3708

North Carolina Electric Membership
Corporation
P. O. Box 27306
Raleigh, North Carolina 27611

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
4830 Concord Road
York, South Carolina 29745

Mr. Virgil R. Autry, Director
Dept of Health and Envir Control
2600 Bull Street
Columbia, South Carolina 29201-1708

Mr. C. Jeffrey Thomas
Manager - Nuclear Regulatory Licensing
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Mr. L. A. Keller
Duke Energy Corporation
526 South Church Street
Charlotte, North Carolina 28201-1006

Saluda River Electric
P. O. Box 929
Laurens, South Carolina 29360

Mr. Peter R. Harden, IV
VP-Customer Relations and Sales
Westinghouse Electric Company
6000 Fairview Road - 12th Floor
Charlotte, North Carolina 28210

Mr. T. Richard Puryear
Owners Group (NCEMC)
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745

Mr. Richard M. Fry, Director
North Carolina Dept of Env, Health, and
Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

County Manager of
Mecklenburg County
720 East Fourth Street
Charlotte, North Carolina 28202

Michael T. Cash
Regulatory Compliance Manager
Duke Energy Corporation

McGuire Nuclear Site
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
12700 Hagers Ferry Road
Huntersville, North Carolina 28078

Dr. John M. Barry
Mecklenburg County
Department of Environmental Protection
700 N. Tryon Street
Charlotte, North Carolina 28202

Mr. Gregory D. Robison
Duke Energy Corporation
Mail Stop EC-12R
526 S. Church Street
Charlotte, NC 28201-1006

Mary Olson
Nuclear Information & Resource Service
Southeast Office
P.O. Box 7586
Asheville, North Carolina 28802

Paul Gunter
Nuclear Information & Resource Service
1424 16th Street NW, Suite 404
Washington, DC 20036

Lou Zeller
Blue Ridge Environmental Defense League
P.O. Box 88
Glendale Springs, North Carolina 28629

Don Moniak
Blue Ridge Environmental Defense League
Aiken Office
P.O. Box 3487
Aiken, South Carolina 29802-3487

Request for Additional Information
McGuire Nuclear Station, Units 1 and 2, and
Catawba Nuclear Station, Units 1 and 2

2.3.1 System Scoping and Screening Results: Reactor Coolant System

2.3.1-1 Borated water leakage through the pressure boundary in pressurized water reactors (PWRs), and resulting borated water induced wastage of carbon steel is a potential aging degradation for the components. Reactor vessel head lifting lugs are considered to be such components requiring aging management. However, if the components are currently covered under Boric Acid Wastage Surveillance Program, then it may not require additional aging management. It appears that the subject components were not discussed in the LRA, and therefore, the staff requests the applicant to verify whether the components are within the surveillance program; and if not, to provide an explanation.

2.3.1-2 Some Westinghouse pressurizers are designed with seismic lugs, and valve support bracket lugs. The staff requests the applicant to verify whether such components exist in McGuire and Catawba plants; and if they do, then to explain why the subject components do not require an aging management review (AMR). Based on past license renewal reviews, the staff believes that the subject components should be within scope requiring aging management, provided the pressurizers are designed with such components.

2.3.1-3 Section 3.9.1.3, page 3.9-4 of McGuire Updated Final Safety Analysis Report (UFSAR), states that the diffuser plate was relied upon when performing the dynamic system load analyses for reactor internals at McGuire to determine the behavior of lower structures when subjected to loads. Furthermore, based on past license renewal reviews of Westinghouse plants, the staff believes that the diffuser plate (provided there is one) should be within the scope requiring aging management because the component provides the safety function of structural and/or functional support for in-scope equipment, and/or provides flow distribution. Please confirm whether the subject component was identified to be within scope requiring aging management for McGuire. If not, explain why. If the UFSAR is incorrect, please indicate if a change to the UFSAR will be made to correct the information.

2.3.1-4 Table 3.1-1 of the LRA identifies components for the steam generators that require AMR. The following components were not listed in the table: anti-vibration bars, stay rod, tube bundle wrapper, and tube support plates. Based on past LRA reviews for the Westinghouse plants, and on the information provided in McGuire and Catawba UFSAR, the staff's view is that these components perform the intended function of providing structural and/or functional support for in-scope equipment, namely the steam generator tubes; and therefore, should be within the scope of license renewal requiring an AMR. If the applicant believes that the intended function of the above components to provide structural and/or functional support for the steam generator tubes is not within the scope of license renewal in accordance with 10 CFR 54.4(a)(2), then the staff requests the applicant to affirm that none of the above mentioned components in McGuire and Catawba units are credited for preventing tube failure during seismic events or during a main steam-line break accident.

2.3.1-5 Catawba drawing CN-1553-1.0, "Flow Diagram of Reactor Coolant System," indicates that piping and components downstream of valve 1NC299 is Duke Class F and is within the scope of the LRA. Catawba drawing CN-2553-1.0, "Flow Diagram of Reactor Coolant System," indicates that piping and components downstream of valve 2NC299 is Duke Class F but is not within the scope of the LRA. Explain why the Unit 2 Duke Class F piping and components of the reactor coolant system are not within the scope of license renewal.

2.3.1-6 McGuire drawing MCFD-2553-02.01, "Flow Diagram of Reactor Coolant System," indicates that valves 2NC0264, 2NC0266, and 2NC0252 and interconnecting piping is Duke Class C but is not within the scope of the LRA. Section 2.1.1.1.1 of the LRA states that Duke Class C piping is within the scope of license renewal. Explain why the Duke Class C piping and components of the reactor coolant system are not within the scope of license renewal.

3.1.1 Reactor Coolant System Class 1 Piping, Valves and Pump Casings

3.1.1-1 Per LRA Table 3.1-1, the loss of material and cracking in orifices are managed by the chemistry control program. Since these restricting orifices are relied upon to separate Class 1 portions from Class 2 portion of the reactor coolant system (RCS) piping in lieu of redundant valves, their continued functionality is extremely important to maintaining the current licensing basis (CLB). It is not evident to the staff how the effectiveness of the chemistry control program to manage loss of material and cracking is verified. No supplemental inservice inspection (ISI) or performance testing is identified. Clarify how the aging effects associated with orifices are adequately managed by the chemistry control program alone, and provide a description of supplemental activities which verify that the chemistry control program is effective.

3.1.2 Pressurizer

3.1.2-1 Section 3.1 of the LRA does not assess whether the potential exists for existing cracks in the pressurizer cladding to grow (as a result of thermal-fatigue induced crack growth) through the cladding and into the ferritic portions of the pressurizer subcomponents that the cladding is joined to. Discuss whether thermal fatigue-induced crack initiation and growth is an issue for the ferritic pressurizer subcomponents that are protected with austenitic stainless steel cladding, and whether propagation of the cracks through the cladding into the ferritic base material or weld material beneath the clad is an applicable effect that requires management. If propagation of the cracks through the cladding into the ferritic base material or weld material beneath the clad is an applicable effect that requires management, state which aging management programs (AMPs) will be used to manage this effect, and justify why you consider the AMPs to be sufficient to manage this effect during the extended periods of operation.

3.1.2-2 The staff is concerned that inter-granular stress corrosion cracking in the heat-affected zones of 304 stainless steel supports that are welded to the pressurizer cladding could grow as a result of thermal fatigue into the adjacent pressure boundary during the license renewal term. The staff considers that these welds will not require aging management in the period of extended operation if the applicant can provide reasonable justification that sensitization has not occurred in these welds during the fabrication of these components. Provide a discussion of how the implementation of plant-specific procedures and quality assurance requirements, if any, for the welding and testing of these austenitic stainless steel components provides reasonable assurance that sensitization has not occurred in these welds and associated heat-affected zones.

3.1.2-3 LRA Table 3.1-1 identifies loss of preload as an aging effect for the manway cover bolts/studs. Table 3.1-1 also indicates that the aging effects associated with the bolts/studs will be managed using the inservice inspection plan and the fluid leak management program. From the description provided in LRA Appendix B for these two AMPs, it is not clear how loss of preload will be managed for the period of extended operation. Clarify how the inservice inspection plan and the fluid leak management program are sufficient to manage loss of preload of the manway cover bolts/studs.

3.1.3 Reactor Vessel and Control Rod Drive Mechanism Pressure Boundary

3.1.3-1 (a) In accordance with LRA Table 3.1-1, aging effects of cracking and loss of material associated with the thimble seal table are managed by the chemistry control program alone. Since mechanical seals between the retractable thimbles and the conduits are provided at the seal table, its continued functionality is extremely important for maintaining the CLB. The staff requests clarification on how the effectiveness of the chemistry control program to manage loss of material and cracking is verified, since no supplemental ISI or performance testing to quantify these effects is identified.

3.1.4 Reactor Vessel Internals

3.1.4-1 In LRA Table 3.1-1, the applicant does not list the rod control cluster assembly guide tube support pins as a separate entry. The staff assumes that they are included with the guide tube assembly. Confirm whether the guide tube support pins at McGuire and Catawba are within the scope of license renewal, and whether the AMRs for the guide tube assemblies in Table 3.1-1 of the application (on pages 3.1-16 and 3.1-17 of the LRA) covers the scope of your AMR for the guide tube support pins. If the guide tube support pins are within scope of license renewal and Table 3.1-1 does not provide an AMR for them, provide an AMR for the guide tube support pins that identifies the aging effects that are applicable to the pins and the aging programs that will be capable of managing the effects.

3.1.4-2 In LRA Table 3.1-1, the applicant did not identify reduction in fracture toughness due to irradiation as one of the applicable aging effects for reactor vessel internal for the lower support plate (forging) and lower core support columns. These materials are fabricated from austenitic stainless steel. In NUREG/CR-6048, Oakridge National Laboratory, on behalf of the NRC, has used $5x10^{20}$ neutrons/cm^2 (E > 1 MeV) as the threshold for loss of fracture toughness due to radiation embrittlement in Type 304 austenitic stainless steel materials. In order to substantiate that loss of fracture toughness is not an applicable effect for these components, confirm that accumulated neutron fluence (E > 1 MeV) for these components during the extended period of operation will be lower than this threshold for radiation induced embrittlement. If the fluence levels for the lower support plate (forging) and lower core support columns are projected to be greater than $5x10^{20}$ neutrons/cm^2 (E > 1 MeV), discuss how you will manage reduction in fracture toughness in these components during the proposed extended periods of operation for the McGuire and Catawba units.

3.1.4-3 In LRA 3.1-1 you list dimensional changes (as a result of radiation-induced void swelling) as an applicable effect for some reactor vessel internal components, but not for others. Confirm that the reactor vessel internal components that you have identified as being potentially susceptible to this effect are the limiting dimensional change (due to void swelling) locations within the reactor vessel cavity, as evaluated from an accumulated neutron fluence basis for the components.

3.1.5 Steam Generator

3.1.5-1 Per Table 3.1-1, the loss of material and cracking in the steam flow limiter, the feedwater thermal sleeves, the handhole diaphragm, and the auxiliary feedwater distribution system are managed by the Chemistry Control Program. No supplemental ISI or performance testing is identified for these SG components. Clarify how the Chemistry Control Program by itself is sufficient to manage loss of material and cracking in these components.

3.1.5-2 In accordance with UFSAR Section 5.4.2.4 for Catawba, the Unit 2 Westinghouse SGs are equipped with a preheater and feedwater flow restrictor with main feedwater delivered just above the tubesheet while the feedwater in the Unit 1 BWI RSGs delivered to the annulus area outside the top of the tube bundle and distributed by a feedring header. It is not clear if the feedwater delivery systems in BWI RSGs at Catawba 1, McGuire 1 and McGuire 2 have flow restrictors.

1. Clarify if the feedwater flow restrictors are present in all four subject plant SG units.

2. Table 3.1-1 identifies the Inservice Inspection Plan and the Chemistry Control Program to detect cracking and loss of material in the flow restrictors and steam flow limiters. Describe the types of inservice inspections performed on these components.

4.2 Reactor Vessel Neutron Embrittlement

4.2-1 In Tables 4.2-1 through 4.2-4 of the application you provide some time-limited aging analyses for upper shelf energies of beltline nozzle plates/forging materials and nozzle weld materials in the McGuire and Catawba vessels. In contrast you did not perform a corresponding pressurized thermal shock assessments for these materials, as would normally be done in Tables 4.2-5 through 4.2-8 of the application. In addition, the staff is not aware that the unirradiated Charpy impact, unirradiated initial RT_{NDT} data (i.e., $RT_{NDT(U)}$ data) and upper shelf energy data and alloying chemistry data (especially copper and nickel contents, as well as phosphorous and sulfur contents) for these nozzle materials have been place on the “dockets” for the McGuire and Catawba reactor units (Dockets 50-369, 50-370, 50-413 and 50-414). With respect to these materials:

1. Submit the corresponding pressurized thermal shock time-limited aging analysis (TLAA) assessments for the nozzle plate/forging materials and nozzle weld materials that were analyzed for upper shelf energy adequacy (as provided for in Tables 4.2-1 through 4.2-4 of the LRA).

2. Submit the unirradiated Charpy impact data, unirradiated initial RT_{NDT} data (i.e., $RT_{NDT(U)}$ data), unirradiated upper shelf energy data, and alloying chemistry data (especially copper and nickel contents, as well as phosphorous and sulfur contents) for the beltline nozzle plates/forging materials and nozzle weld materials in the McGuire and Catawba vessels on the respective dockets for the McGuire and Catawba reactor units (i.e., Dockets Nos. 50-369, 50-370, 50-413 and 50-414). Provide your bases for the data being docketed.

4.3 Metal Fatigue

4.3-1 Section 4.3.1 of the LRA discusses the Duke evaluation of the fatigue TLAA for ASME Class 1 components. The discussion indicates that Duke will rely on its Thermal Fatigue Management Program (TFMP) to assure that component fatigue evaluations remain valid for the period of extended operation. Tables 5-2 and 5-49 of the of the McGuire UFSAR and Table 3-50 of the Catawba UFSAR contain a list transient design conditions and associated design cycles. Provide the following information for each transient listed in these tables:

1. The current number of operating cycles and a description of the method used to determine the number and severity of the design transients from the plant operating history.

2. The number of operating cycles estimated for 60 years of plant operation and a description of the method used to estimate the number of cycles at 60 years.

4.3-2 The Westinghouse Owners Group issued Topical Report WCAP-14577, Revision 1-A, "Aging Management for Reactor Internals," to address the aging management of the reactor vessel internals (RVI). The staff review of WCAP-14577, Revision 1-A identified a number of issues that should be addressed on a plant specific basis. Renewal Applicant Action Item 11 specified in WCAP -14577, Revision 1-A indicates that the fatigue TLAA of the reactor vessel internals should be addressed on a plant specific basis. In the LRA, Duke indicates that the TFMP will assure that component fatigue analyses will remain within their design values for the period of extended operation. List the transients that contribute to the fatigue usage for each component listed in Table 3-3 of WCAP-14577, Revision 1-A and discuss how the TFMP monitors these transients.

4.3-3 The Westinghouse Owners Group issued Topical Report WCAP-14575-A, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components," to address aging management of the RCS piping. Tables 3-2 through 3-16 of WCAP-14575-A list RCS components where fatigue is considered significant. The staff review of WCAP-14575-A identified a number of issues that should be addressed on a plant specific basis. Renewal Applicant Action Item 8 requests that the applicant to address components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575-A. Duke indicates that the TFMP will assure that component fatigue analyses will remain within their design values for the period of extended operation. Discuss how the TFMP addresses the components labeled I-M and I-RA in Tables 3-2 through 3-16 of WCAP-14575-A.

4.3-4 The Westinghouse Owners Group has issued the generic Topical Report WCAP-14574-A to address aging management of pressurizers. The staff review of WCAP-14574-A identified a number of issues that should be addressed on a plant specific basis. Renewal Applicant Action Item 1 requests that the applicant demonstrate that the pressurizer sub-component cumulative usage factors (CUFs) remain below 1.0 for the period of extended operation. Table 2-10 of WCAP-14574-A indicates that the ASME Section III Class 1 fatigue CUF criterion could be exceeded at several pressurizer sub-component locations during the period of extended operation. WCAP-14574-A also identified recent unanticipated transients that were not considered in the original ASME Section III Class 1 fatigue analyses, including inflow/outflow thermal transients. Provide the following information:

1. Confirm that the additional transients discussed in WCAP-14574-A, not considered in the original design, have been addressed at McGuire and Catawba.

2. Show the ASME Section III Class 1 CLB CUFs for the applicable sub-components of the McGuire and Catawba pressurizers specified in Table 2-10 of WCAP-14574-A and the corresponding CUFs for the extended period of operation.

3. Discuss the impact of the environmental fatigue correlations provided in NUREG/CR-6583, "Effects of LWR [Light Water Reactor] Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," on the above results.

4.3-5 Section 4.3.1.2 of the LRA discusses Duke's evaluation of the impact of the reactor water environment on the fatigue life of components. The discussion indicates that Duke's evaluation will use method 2 contained in draft Electric Power Research institute (EPRI) report, "Guidelines for Addressing Fatigue Environmental Effects in a License Renewal Application." The evaluation will address the fatigue sensitive component locations identified in NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." Provide the following additional information regarding the evaluation of reactor water environmental effects:

1. Confirm that the environmental fatigue correlations contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue on Fatigue Design Curves of Austenitic Stainless Steels," will be used in the evaluation.

2. Provide the design basis usage factors for each of the six component locations listed in NUREG/CR-6260.

3. Note 1 of the Duke procedure indicates that ASME Section XI flaw tolerance and inspection procedures may be used as an alternative method to manage environmental fatigue. The NRC staff has not endorsed a procedure on a generic basis which allows for ASME Section XI inspections in lieu of meeting the fatigue usage criteria. Duke has not provided a technical basis demonstrating the technical adequacy of its proposal. Provide a detailed technical evaluation which demonstrates the proposed inspections

provide an adequate technical basis for detecting fatigue cracking before such cracking leads to through wall cracking or pipe failure. The detailed technical evaluation should be sufficiently conservative to address all uncertainties associated with the technical evaluation (e.g., fatigue crack initiation and detection, fatigue crack size, and fatigue crack growth rate considering environmental factors). As an alternative to the detailed technical evaluation, provide a commitment monitor the fatigue usage, including environmental effects, during the period of extended operation, and to take corrective actions, as approved by the staff, if the usage is projected to exceed one.

4. Note 2 of the Duke procedure indicates that the environmental factor will be adjusted to by a Z factor to take credit for moderate environmental effects in the existing ASME fatigue curves. The staff considers the use of the Z factor an open issue regarding implementation of the EPRI procedure (Meeting summary dated March 1, 2001). Provide additional data and additional data evaluations that demonstrate (1) there is sufficient margin in the procedure to account for material variability and experimental data scatter, size effects, surface finish effects and loading history, (2) that environmental effects and surface effects are not independent effects. As an alternative, revise the Duke procedure to eliminate the use of the Z factor.

4.3-6 The LRA does not address the issue of underclad cracks. The Westinghouse Owners Group (WOG) submitted for staff review topical report WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants (MUHP-6110)" by letter dated March 1, 2001. This report describes the fracture mechanics analysis that evaluates the impact of 60 years of operation on reactor vessel underclad crack growth and reactor vessel integrity. However, in a letter dated April 12, 2001, the staff identified area where additional information is needed to complete its review of WCAP-15338. The WOG response to the RAI is contained in letters dated June 15, 2001, and July 31, 2001. The WOG response indicates that the pressurized thermal shock portion of the analysis applies to three loop Westinghouse plants. WCAP-15338 indicates that underclad cracks are confined to forging materials, SA 508 Class 2 and 3. WCAP-15338 also indicates that underclad cracks were observed in SA 508 Class 3 nozzles clad with multiple-layer, strip electrode, submerged-arc welding processes where preheating and post-heating were applied to the first layer but not to the subsequent layers. Provide the following information:

1. Identify any reactor vessel components that were fabricated from SA 508 Class 2 or 3 forgings.

2. Indicate whether any of the SA 508 Class 2 or 3 forgings identified above are susceptible to underclad cracking.

3. Indicate whether any of the SA 508 Class 2 or 3 forgings are subject to neutron embrittlement (i.e., subject to a neutron fluence greater than or equal to 10^{17} n/cm^2 [E>1MeV]).

4. If any forgings are susceptible to underclad cracking, identify the basis for concluding that the cracks will not result in loss of reactor vessel integrity during the period of extended operation. The assessment should consider the impact of fatigue and neutron embrittlement on the underclad cracks.

4.3-7 Section 4.3.2 of the LRA addresses ASME Section III, Class 2 and 3 piping fatigue. The LRA indicates that two locations at McGuire and Catawba could reach the 7,000 cycle limit during the period of extended operation. Identify these locations and indicate how the number of expected cycles was determined. Also describe the re-evaluation that was performed to demonstrate these locations will be acceptable for the period of extended operation.

4.7.1 Reactor Coolant Pump Flywheel Fatigue

4.7.1-1 Section 4.7.1 of the LRA discusses the analysis related to a 60-year fatigue life for the reactor coolant pump fly wheel. Provide a summary of the existing design basis analysis to enable the staff to evaluate the validity of fatigue life for the extended period of operation.

B.3.1 Alloy 600 Review

B.3.1-1 Confirm that the following aspects of your Alloy 600 Review are valid:

1. The Alloy 600 Review is simply a susceptibility ranking review calculation that will be used to determine whether inspection techniques proposed in aging management programs for managing aging effects in Alloy 600 components of the reactor coolant pressure boundary components (including reactor vessel internal components) should be enhanced or augmented.

2. The program attributes are normally provided in the application for programs that are listed in the LRA as aging management programs. Since the Alloy 600 Review is simply a review program, the program attributes for the review are not necessary.

B.3.5 Bottom-Mounted Instrumentation Thimble Tube Inspection Program

B.3.5-1 For Catawba Unit 1, the applicant had performed thimble inspections in 1988, 1993, and 1999; for Catawba Unit 2, the applicant had performed inspections in 1989, 1990, and 1993. No significant changes in wear rates were detected in both units, and no tubes are capped in Unit 1; however, two tubes are capped in Unit 2 due to wear concerns. The LRA indicates that no further testing will be required until 2008 for Unit 1, and 2007 for Unit 2. Based on the description in the LRA, it appears that tube wear condition is more severe in Unit 2 than Unit 1. Explain why the projected next testing for Unit 2 (in 2007) is fourteen years after the previous testing (in 1993) in comparison with nine years (1999 to 2008) in Unit 1, and provide details of wear projection calculations of both units. Are the thimble tubes designed similarly (such as same tube wall thickness) for both units? Or is there a modified design that is used in Unit 2? What is the allowable number of thimbles that may be capped? Will the allowable capped number be exceeded for extended plant operation of 20 more years? Should this happen, what corrective actions will be taken?

B.3.5-2 For McGuire, the LRA indicates that the Unit 1 thimble inspections had been performed in 1988 and 2001 with 10 tubes showing detectable wall loss. Two tubes were capped due to other types of damage. The Unit 2 inspections had been performed in 1989 and 1993 with eight tubes showing wear. The future inspections are planned in 2008 for

Unit 1 and in 2005 for Unit 2. Clarify the type of “other damage” in the two capped tubes at Unit 1, and provide more details of tube wear projection calculations at both units. Are tubes with modified design being used in either units?

B.3.5-3 Since a thimble tube failure will result in leakage of reactor coolant, verify whether a leaking thimble tube can be isolated, and describe the corrective actions to be taken under such circumstances.

B.3.5-4 The design of thimble tubes has evolved with respect to their thickness, gap size between tube wall and guide tube, and isolation techniques since the issuance of Inspection and Enforcement (IE) Bulletin 88-09. In order to demonstrate that continued implementation of the existing thimble tube inspection program is capable of monitoring tube wall thinning prior to loss of component intended function during the extended plant operation for 20 more years, supplement the summary of industry experience regarding the performance of bottom-mounted instrumentation thimble tubes, specifically with the same design, if any, as the tubes used in McGuire and Catawba.

B.3.9 Control Rod Drive Mechanism (CRDM) Nozzle and Other Vessel Closure Penetrations Inspection Program

B.3.9-1 The CRDM Nozzle and Other Vessel Closure Penetration Inspection Program, described in Section B.3.9 of Appendix B the LRA, is designed to manage cracking in the Alloy 600 vessel head penetration (VHP) nozzles of the McGuire and Catawba units. In Section B.3.9 of the LRA, the applicant did not specify whether it would continue to be a participant in the NEI program for managing primary water stress corrosion cracking (PWSCC) type aging in Alloy 600 VHP nozzles of U.S. pressurized water reactor (PWR) designed facilities, and whether the applicant would continue to use the program as a basis for evaluating the Alloy 600 VHPs in the McGuire and Catawba nuclear units during the proposed extended operating terms for the units. With respect to this program:

1. Discuss how the recent circumferential cracking discussed in NRC Bulletin 2001-01 will impact your management program for the McGuire and Catawba CRDM penetration nozzles and other vessel head penetration nozzles.

2. Discuss what additional activities you will be participating in, if any, that will be implemented as part of this program.

B.3.26 Reactor Vessel Integrity Program

B.3.26-1 For Catawba Unit 1: In Table B.3.26-2, the staff notes that the 32 effective full power years (EFPY) inner diameter (ID) vessel fluence is 2.334 (in terms of 10^{19} n/cm^2). However, the projected value in WCAP-11527, Table 6-11, is 3.17 (no azimuth is specified), in WCAP-13720 Table 6-17, at 25° is 2.52 and in WCAP-15117, Table 6-14 for 34 EFPYs is 1.98 at 30°. The updating of the older values should have resulted in higher values. Please explain the apparent discrepancies (projected low leakage loadings) and the physics of the updating which justifies the differences. Why does the maximum occur at slightly different azimuths?

B.3.26-2 For Catawba Unit 2: Same as with Catawba Unit 1 (see previous RAI, B.3.26-1), regarding reported values for 32 EFPYs in WCAP-11941 and WCAP-13875 vs the submittal Table B.3.26-2. In addition WCAP-13875 does not report calculated values. (Note: in WCAP-11941, Table 6-13 values at 25°, 30° and 45°. Is there a typo? The maximum should be at 25°). Please explain the apparent discrepancies (projected low leakage loadings) and the physics of the updating which justifies the differences. Why does the maximum occur at slightly different azimuths?

B.3.26-3 For McGuire Unit 1: In Table B.3.26-1 of the submittal, a 54 EFPY fluence is reported and referenced to WCAP-14993 which does not include 54 EFPY values. Please explain. The 1/4T value for 32 EFPYs reported in WCAP-12354 is significantly different than the value reported in the submittal and referenced to WCAP-12354. Please explain. The value reported in WCAP-10786 for 32 EFPYs and the corresponding value reported in the submittal and referenced to WCAP-13949 are significantly different. Please explain. A fluence value is reported in Table B.3.26-1 of the submittal for 54 EFPY of the vessel and referenced to WCAP-14993, which does not report values at 54 EFPYs. In addition the value reported for 50.3 EFPY is almost the same. Please explain.

B.3.26-4 For McGuire Unit 2: Table B.3.26-1 of the submittal reports a 54 EFPY value at 1/4T referenced to WCAP-13516 which does not report values above 32 EFPY. How was that value derived? The 1/4T, 32 EFPY value reported in the same table and referenced to WCAP-12556 does not agree with the value reported in the table. Please explain. The 54 EFPY ID value reported in the same table and referenced to WCAP-14799 does not exist in WCAP-14799 which does not report 54 EFPY values. Please explain why. The 32 EFPY ID value reported in the same table was calculated with END/B-IV cross sections in WCAP-13516. In addition justify why this value not reevaluated, especially when the location of the surveillance capsule is behind the neutron pad?

<u>B.3.27 Reactor Vessel Internals Inspection Program</u>

B.3.27-1 The applicant has identified change in dimensions due to void swelling as an applicable aging effect; it will be managed by the Reactor Vessel Internals Inspection. In Section B.3.27 “Monitoring and Trending”, the applicant states that McGuire and Catawba will rely upon the results of the inspections at Oconee to assess the effects of void swelling. It is not clear to the staff whether the Oconee results will be applicable to McGuire and Catawba, because the RVI components are of different designs (B&W vs. Westinghouse), may utilize different materials of construction, and may be subject to different fluence rates. Provide additional information that supports the technical validity of this extrapolation, specifically addressing the similarities and differences pertaining to RVI design details; materials of construction; reactor power rating and neutron fluence levels; and critical locations where dimensional changes may compromise performance of intended functions.