Duke Power®
A Duke Energy Company

Duke Energy Corporation
526 South Church Street
PO Box 1006
Charlotte, NC 28201-1006
(704) 382-2200 OFFICE
(704) 382-4360 FAX

Michael S. Tuckman
Executive Vice President
Nuclear Generation

September 6, 2002

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington D.C. 20555

Subject: Duke Energy Corporation
Oconee Nuclear Stations Units 1, 2 & 3
Docket Nos. 50-269, 270, 287

30 Day Response to NRC Bulletin 2002-02:
Reactor Pressure Vessel Head and Vessel Head
Penetration Nozzle Inspection Programs

Attached to this letter, pursuant to the requirements of 10 CFR 50.54(f) is an Enclosure which constitutes Duke Energy (Duke) Corporation's 30 day response to NRC Bulletin 2002-02 for all three units at Oconee Nuclear Station. The response specifically addresses Bulletin item 1.A listed in Enclosure I of the subject Bulletin. Duke Energy has entered into an agreement with Babcock and Wilcox to replace the vessel heads at all three units. Manufacturing of the heads began in the spring of 2001 and final delivery of the last head is expected in the spring of 2004.

Duke has scheduled replacement of the Reactor Pressure Vessel Heads (RPVHs) for Oconee Units 1 and 3 following the current operating cycles. No RPVH inspections are planned for these units prior to RPVH replacement. The replacement heads incorporate Alloy 690 nozzles and are expected to be resistant to primary water stress corrosion cracking experienced in the Alloy 600 nozzles installed in the current heads. Inspection plans for the replacement RPVHs will be developed in response to industry operating experience. Duke will also respond to relevant information developed by the Electric Power Research Institute or the American Society of Mechanical Engineers.

Unit 2 at Oconee Nuclear Station is scheduled to begin the last refueling outage prior to head replacement in mid October 2002. The established inspection plan for this outage follows the graded approach developed at Oconee over the last several years. This graded approach has been previously described to the Nuclear Regulatory Commission (NRC) Staff in responses to previous bulletins.[1] Duke also plans a supplemental 100% volumetric blade probe inspection of

[1] Letter from W R McCollum Jr to NRC, Response to NRC Bulletin 2001-01: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, Dated August 28, 2001 and
Letter from K S Canady to NRC, Response to NRC Bulletin 2002-01: Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity , Dated April 1, 2002

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the previously un-repaired nozzles. The Unit 2 RPVH is scheduled to be replaced in the spring of 2004 following the next operating cycle.

Duke Energy has collected substantial amounts of information regarding the condition of the Oconee heads and nozzles since the emergence of this issue in the early 1990's. This wealth of information combined with the inspection techniques developed at Oconee allow for accurate assessments of current and future vessel head integrity. The attached bulletin response provides details regarding past history, a description of the established and supplemental inspection plans, a description of plans for additional monitoring and justification for the sufficiency of these actions.

It is Duke Energy's position that the established graded inspection plan alone provides a sufficient basis for continued safe operation and demonstration of ongoing compliance with all regulatory requirements. As a conservative measure, Duke has scheduled a volumetric blade probe inspection of the previously un-repaired nozzles during the upcoming Unit 2 refueling outage. As an additional conservative measure, Duke will implement routine reviews of certain plant data which is expected[2] to provide indirect indication of ongoing vessel head wastage.

Duke's efforts and investment in investigation and resolution of this issue are well documented and have been substantial. This investigative effort led to Duke's decision to purchase replacement heads early in 2001. Duke requests that the NRC consider this prior experience as well as the conservative course of action described in the attached response when reviewing this matter. Since the upcoming Unit 2 refueling outage is rapidly approaching, an expedited staff review would be beneficial for business planning purposes. This information was collected using reasonably available sources and means available to meet the requested 30 day response. If you have questions or need additional information, please contact Gregory S. Kent at (704)373-6032.

M. S. Tuckman

M. S. Tuckman

ENCLOSURES

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[2] NRC Information Notice 2002-13 "Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradation

xc: L.A. Reyes
U.S. Nuclear Regulatory Commission
Regional Administrator,
Region II Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

L.N. Olshan
NRC Project Manager (ONS)
U.S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

M.C. Shannon
Senior Resident Inspector (ONS)

M. S. Tuckman affirms that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth herein are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to me: September 6, 2002
Date

Michael T. Cash
Notary Public

My Commission Expires: January 22, 2003
Date

SEAL

MICHAEL T. CASH
Notary Public
Lincoln County, North Carolina
Commission Expires January 22, 2003

bcc: L.F. Vaughn – PB05E
M.T. Cash – EC05O
L.E. Nicholson – ON03RC
M.R. Robinson – EC09C
G.S. Kent – EC05O
J.M. Shuping – EC09O
D.E. Whitaker – EC09O
J.E. Smith – ON03RC
Oconee Master File – ON03DM
ELL – EC05O

Response Summary and Overview

Inspections and Monitoring

Duke will replace the Reactor Pressure Vessel Heads (RPVHs) on Oconee Units 1 and 3 following the current operating cycles. The RPVH replacement refueling outage will occur during the EOC-21 RFO scheduled begin in 13 months for Unit 1 and during the EOC-20 RFO scheduled begin in about eight months for Unit 3. No RPVH inspections are planned for these units prior to RPVH replacement. The replacement heads incorporate Alloy 690 nozzles and are expected to be resistant to primary water stress corrosion cracking experienced in the Alloy 600 nozzles installed in the current heads. Inspection plans for the replacement RPVHs will be developed in response to industry operating experience. Duke will also respond to relevant information developed by the Electric Power Research Institute or the American Society of Mechanical Engineers.

Unit 2 at Oconee Nuclear Station is scheduled to begin the last refueling outage prior to head replacement in mid October 2002. The established inspection plan for this outage follows the graded approach developed at Oconee over the last several years. This graded approach has been previously described to the Nuclear Regulatory Commission (NRC) Staff in responses to previous bulletins.[1] Duke also plans a supplemental volumetric blade probe inspection of the previously un-repaired nozzles. The Unit 2 RPVH is scheduled to be replaced in the spring of 2004 following the next operating cycle.

In addition, the Unit 2 inspection will include an examination referred to as "leak path technology" which uses signals corresponding to the back-wall of the nozzle to investigate the integrity of the nozzle to shell shrink-fit area. The signals will be provided by the same transducers used to inspect for base metal flaws. This leak path technology will be used to evaluate each nozzle that is subjected to UT blade probe inspection. Plots of the "C" scan amplitude at the nozzle back-wall will be examined to ensure the integrity of the shrink-fit is good. By confirming the presence of contact throughout the shrink fit zone, ONS can conclude no through wall leaks and cracks exist that could pose a safety issue. Additional inspection of the weld J-groove by dye penetrant methods will be performed for any nozzle where the results of the leak path technology evaluations are inconclusive.

As an additional conservative action, Duke will implement a number of enhancements that will improve the capability for identifying symptoms of RPVH degradation which could occur during operation. These enhancements include provisions focused on indications of reactor coolant system (RCS) leakage and metal wastage. These enhancements address the issues described in NRC Information Notice 2002-13 "Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradation".

[1] Letter from W R McCollum Jr to NRC, Response to NRC Bulletin 2001-01: Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, Dated August 28, 2001 and
Letter from K S Canady to NRC, Response to NRC Bulletin 2002-01: Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity, Dated April 1, 2002

Justification

Oconee Units 1 and 3 are in the final cycles of operation before scheduled replacement of the Reactor Pressure Vessel Heads (RPVHs). Duke Energy has collected substantial amounts of information regarding the condition of the Oconee heads and nozzles since the emergence of this issue in the early 1990's. This wealth of information combined with the inspection techniques developed at Oconee allow for accurate assessments of current and future vessel head integrity. This bulletin response provides details regarding past history, a description of the established and supplemental inspection plans, a description of plans for additional monitoring and justification for the sufficiency of these actions. From November 2000 until present day, Duke has continued to improve inspection techniques, interpretation of inspection results, documentation, and RPVH cleaning. These improvements allow visual inspections to be used as the primary method for successfully detecting leakage of the Alloy 600 nozzles. For Oconee Unit 2 these techniques will be augmented with UT examination, "leak path technology" and operational monitoring.

It has been established that circumferential cracks will be preceded by through wall axial cracks which will result in visible leakage. Primary water leakage from through wall axial cracks must be present to initiate a circumferential crack. Circumferential cracks can only initiate after primary water leakage into the annulus. Therefore, any circumferential cracks that occur during the current operating cycle will be detected during visual inspections during the upcoming refueling outage. As described above, circumferential cracks that may initiate during the upcoming operating cycle will not have sufficient time to grow to a size that could compromise code limits and thus avoid the potential of a rod ejection. Visual inspections performed during the upcoming Unit 2 refueling outage provide a reasonable basis for concluding that circumferential cracks which could challenge code limits will not occur during the final operating cycle before head replacement.

The PWSCC observed at Oconee has consistently manifested itself as small deposits of boron characteristic of weepage. Oconee has never observed large accumulations of boron as previously thought to occur with higher leakage. This discovery led Duke to refine visual methods for leak identification and to improve. No substantial wastage of carbon steel has been observed during the investigation of any leakage on the ONS RPVHs. Further, in the course of completing repairs of the ONS nozzles by manual and automatic means, up-close and detailed inspections have been completed of the head and bore area and no evidence of significant wastage has been found. This operational experience combined with the planned inspections and monitoring provide a sound basis for concluding that consequential head wastage will not occur at Oconee.

To summarize, the ONS-2 RPVH inspection plan is a graded or stepped approach that utilizes the following:

- 100% qualified visual inspection of all CRDM nozzles.

- Blade probe volumetric inspection of all remaining un-repaired nozzles to locate both ID and OD initiated axial and circumferential cracks and potential leak pathways.
- Dye penetrant inspection of indeterminate blade probe UT inspection leak path results.

Duke's confidence in this graded approach is based upon the above technical justification in addition to our previous efforts and experience in addressing these concerns. This confidence is further strengthen by additional ONS-2 specific analysis that determined the core damage frequency associate with this situation is well below Regulatory Guide 1.174 threshold valves and that no fuel rod failures would occur given the unlikely occurrence of a rod ejection accident

ENCLOSURE I
Oconee Nuclear Station
Response to NRC Bulletin 2002-02

Requested Information

1. Within 30 days of the date of this bulletin:

A. PWR addressees who plan to supplement their inspection programs with non-visual NDE methods are requested to provide a summary discussion of the supplemental inspections to be implemented. The summary discussion should include EDY, methods, scope, coverage, frequencies, qualification requirements, and acceptance criteria.

B. PWR addressees who do not plan to supplement their inspection programs with non-visual NDE methods are requested to provide a justification for continued reliance on visual examinations as the primary method to detect degradation (i.e., cracking, leakage, or wastage). In your justification, include a discussion that addresses the reliability and effectiveness of the inspections to ensure that all regulatory and technical specification requirements are met during the operating cycle, and that addresses the six concerns identified in the Discussion Section of this bulletin. Also, include in your justification a discussion of your basis for concluding that unacceptable vessel head wastage will not occur between inspection cycles that rely on qualified visual inspections. You should provide all applicable data to support your understanding of the wastage phenomenon and wastage rates.

Response for Oconee Units 1 and 3

Duke will replace the Reactor Pressure Vessel Heads (RPVHs) on Oconee Units 1 and 3 following the current operating cycles. No RPVH inspections are planned for these units prior to RPVH replacement. Inspection plans for the new RPVHs will consider industry operating experience and work by the Electric Power Research Institute (EPRI) Material Reliability Program (MRP) and work ongoing in the American Society of Mechanical Engineers (ASME) Code.

Oconee Nuclear Station Unit 1 [23.23 Effective Degradation Years (EDY) at the next scheduled Refueling Outage (RFO)] is currently into its last operating cycle before the existing RPVH is to be replaced in the fall of 2003. The RPVH replacement refueling outage will occur during the End-of-Cycle (EOC) 21 RFO scheduled to begin in 13 months. Oconee Nuclear Station Unit 3 (22.47 EDY at the next scheduled RFO) is currently into its last operating cycle before the existing RPVH is to be replaced in the spring of 2003. The RPVH replacement refueling outage will occur during the EOC-20 RFO scheduled to begin in about eight months.

Response for Oconee Unit 2

Oconee Nuclear Station Unit 2 (ONS-2) is scheduled to begin its next refueling outage in October 2002 (23.70 EDY at the next scheduled RFO). The RPVH inspection plan for Unit 2 will be consistent with Oconee Nuclear Station's response to NRC Bulletin 2001-01, 2002-01 and methods used to manage past RPVH inspections and repairs. This plan will also be supplemented with a volumetric inspection of the sixty-five un-repaired Control Rod Drive Mechanisms (CRDM) nozzles. At the conclusion of the October refueling outage, Unit 2 will begin its final cycle of operation before the present RPVH is replaced. The Unit 2 RPVH is scheduled to be replaced in the spring of 2004 or about 16 months following the October 2002 outage.

Inspection Plan for ONS-2

The ONS-2 head was cleaned during the last RFO in preparation for the final qualified visual examination prior to RPVH replacement. The RPVH inspection plan is a graded approach starting with a 100% bare metal visual to inspect a full 360° around each of the 69 CRDM nozzles. Any evidence of leakage, head wastage or any nozzle(s) that may have masked deposits of boron or other matter around the nozzle will be noted and investigated further. Deposits will be removed and the previously masked area visually inspected for wastage. Additional visual inspections of the area inside the service structure above the insulation such as inspections for flange leakage will be used to aid in the disposition of a suspect nozzle. The bare metal visual inspection of the RPVH will be performed and documented in accordance with written procedures and acceptance criteria. Inspections are conducted by an experienced component engineer and a Quality Assurance (QA) inspector qualified to perform VT-2 inspections. The visual acceptance standards and supplemental actions required when evidence of leaks is found meet the intent of ASME Section XI. (VT-2 inspections are normally performed at normal operating temperature and pressure.)

Nozzles showing signs of leakage or any nozzle(s) that may have a masking deposit of boron or other deposit around the nozzle will be subjected to ultrasonic (UT) volumetric inspections. UT will be performed by qualified personnel using procedures and equipment demonstrated on EPRI blind mockup test blocks witnessed by the NRC in August 2002. Penetrant testing (PT) will be performed to detect possible cracking in the J-groove weld (weld) if the UT inspection results do not identify a through wall indication (or "leak path") on a masked nozzle.

As a conservative action, Duke will volumetrically inspect the remaining sixty-five, un-repaired, Unit 2 nozzles using blade probe ultrasonic technology that is delivered by an automated inspection tool from under the RPVH. Blade probe UT technology is a recent development for the inspection of RPVH penetrations and was successfully used during the ONS-3 outage in November 2001. This inspection method has the benefit of allowing volumetric inspection of the nozzle base material without having to remove the control rod drive for inspection access. The blade probe UT inspection will be conducted by qualified personnel using equipment and procedures, with acceptance criteria, demonstrated on EPRI blind mockup test blocks witnessed by the NRC in August 2002. Blade probe UT utilizes time of flight, tip diffraction techniques (TOFD), optimized for circumferentially oriented cracks. This blade probe technology has demonstrated the ability to detect axial cracks in the nozzle base metal, identifying both OD and ID initiated axial and circumferential indications. Blade probe scanning is effective axially to 11.25 inches which is sufficient to examine most nozzles from the bottom edge to the top of the head.

In addition, an examination commonly referred to as "leak path" will utilize signals corresponding to the back-wall of the nozzle to investigate the integrity of the nozzle to shell shrink-fit area. The signals will be provided by the same transducers used to inspect for base metal flaws. Data shows that the shrink fit used in construction of most RPVHs is sufficiently tight to transmit measurable portions of the sound energy through the contacting interface while the majority of the energy is reflected. The difference in the energy level reflected off the back-wall between areas where the fit is tight, and areas where contact is lost due to corrosion, is utilized to investigate the integrity of the nozzle shrink-fit and can accurately define an area where leakage may have occurred.

Investigations have shown that in nozzles where leakage has occurred the ultrasonic amplitude of back-wall response at the nozzle OD increases, which easily shows up in a "C" scan map of the shrink-fit area. A "C" scan map of the ultrasonic data represents the circumferential-axial plane with the depth data compressed. A leak path shows up as a river like pattern that outlines the slight loss of material and resulting loss of contact. The amplitude plots are not capable of determining the depth of any lost material. Leakage, either through the nozzle or through the weld, must eventually pass through the root of the "J" groove weld and a point where the buttering, vessel head base material, and nozzle base material join. This point is commonly referred to as the triple point. The shrink fit contact area begins just above the triple point and extends for several inches (Reference Sketch 1). Leakage through the weld material alone must also pass through the triple point and into the shrink-fit area. Once it passes into the shrink fit area the slightest amount of corrosion of the carbon steel head will reduce the shrink-fit contact area and will be detected by the leak path methods. Leak path methods are effective in ensuring that the weld material does not contain any through weld pressure boundary leaks and that leakage is not occurring on the nozzle being investigated.

Leak path technology will be used to evaluate each nozzle that is subjected to UT blade probe inspection. Plots of the "C" scan amplitude at the nozzle back-wall will be examined to ensure the integrity of the shrink-fit is good. By confirming the presence of contact throughout the shrink fit zone, ONS can conclude no through wall leaks and cracks exist that could pose a safety issue. Additional inspection of the weld J-groove by dye penetrant methods will be performed for any nozzle where the results of the leak path technology evaluations are inconclusive.

To summarize, the ONS-2 RPVH inspection plan is a graded or stepped approach that utilizes the following:

- 100% qualified visual inspection of all CRDM nozzles.
- Blade probe volumetric inspection of all remaining un-repaired nozzles to locate both ID and OD initiated axial and circumferential cracks and potential leak pathways.
- Dye penetrant inspection of indeterminate blade probe UT inspection leak path results.

Justification for ONS-2 Inspection Plan

Background

Cracking of the RPVH head was initially discovered at Unit 3 of the EdF Bugey plant in 1991. The cracking at Bugey 3 was determined to be due to Pressurized Water Stress Corrosion Cracking (PWSCC) that initiated on the ID of the Alloy 600 CRDM penetration housings. PWSCC cracking of other components fabricated from Alloy 600 had occurred. Degradation due to PWSCC of the RPVH penetrations was anticipated by Duke as a result of industry experience and root cause work performed to determine the cause of industry leakage events.

Duke implemented a long term strategy to prepare for possible cracking as a result of the Bugey 3 experience. Duke's actions reflected the philosophy of an aggressive approach focusing on early identification. Oconee re-analyzed and modified the reactor head service structures to create access ports to facilitate inspection and cleaning of the RPVHs. ONS-2 was the lead B&W plant for PWSSC susceptibility and a 100% Eddy Current Test (ECT) inspection of the inside diameter (ID) of the penetration tubes was conducted in 1994. ECT inspections were conducted on a sample of nozzles in 1996 and 1999. The 1994 inspection indicated only shallow ID craze cracks. Analysis of the inspection results from subsequent inspections showed no crack growth from 1994 to 1999. (Refer to NRC Bulletin 2001-01 response for additional details.)

The service structure modification allowed the initial cleaning of RPVHs to remove accumulated boron. The head was generally clean and in such condition that boric acid leaks characteristic of those predicted in "Safety Evaluation for B&W Designed Reactor Vessel Head Control Rod Drive Mechanism Nozzle Cracking", May 1993 would have been identified.

In November 2000, during a routine visual inspection of the Unit 1 RPVH, Oconee personnel identified boric acid deposits around a single CRDM nozzle and several thermocouple nozzles. Further inspection ruled out CRDM flange leakage as the source of the observed boron deposits. Supplemental NDE and metallurgical analysis performed as part of a thorough root cause investigation attributed the cracking and leakage to PWSCC. The leakage manifested itself as small deposits of boron characteristic of weepage instead of large accumulations of boron as previously thought to occur with higher leakage. This discovery led Duke to refine visual methods for leak identification and to improve head cleaning. Improved inspection procedures were formalized assuring that all the CRDM nozzles were inspected systematically and the results documented.

From November 2000 until present day, Duke has continued to improve inspection techniques, interpretation of inspection results, documentation, and RPVH cleaning. These improvements allowed visual inspections to become the primary method for successfully detecting leakage of the Alloy 600 nozzles. This approach has allowed Duke to detect and repair leaking nozzles prior to the occurrence of unacceptable wastage of the carbon steel RPVH due to boric acid corrosion. Visual inspection has successfully detected 20 leaking nozzles between the three Oconee units. Cracks have been confirmed by supplemental NDE for all nozzles visually identified as leaking. Seventy-three (73) nozzles from the three Oconee units were also examined by NDE for extent of condition and found to be clear of any axial or circumferential indications (with the exception of superficial ID crazed cracking). Thus, ONS and other industry experience provide a sound basis for the use of visual inspections to maintain integrity of the ONS RPVHs.

Duke strongly believes that the graded RPVH inspection plan described above and in response to other NRC related bulletins (2001-01 and 2002-01) provides assurance that the ONS units are structurally sound and that all regulatory commitments are being met. However, as additional conservative actions, Duke will conduct a volumetric inspection of the remaining previously unrepaired nozzles using blade probe ultrasonic technology as previously discussed and incorporate lessons learned from the Davis Besse root cause report (ref. 7) to evaluate conditions in containment for signs of leakage/wastage. These additional actions demonstrate Duke's commitment to safe plant operation and role as an industry leader.

Basis for Continued Safe Operation

Design Features and Inspection

ONS-2 CRDM nozzles are robust in design to satisfy all applicable design requirements. The components were fabricated using Alloy 600, which has excellent general corrosion resistance and extremely high fracture toughness. The penetration design allows for visual, surface, and volumetric inspection.

The approach for preventing degradation of the ONS-2 RPVH during the last cycle prior to head replacement starts with a 100% bare metal qualified visual inspection of the RPVH. Previous ONS inspection history demonstrates that a qualified visual inspection provides an adequate means of detecting leakage thus preventing both unacceptable wastage of the head and rod ejection.

The ONS-2 inspection plan is consistent with the previous five ONS RPVH inspections and includes an additional conservative action, the performance of blade probe UT of the remaining previously unrepaired nozzles. This plan is based on the observation that leakage through the pressure boundary at Oconee shows up as leakage around the annulus of the nozzles. This approach has been validated through industry experience with qualified visual inspection and by analytical evaluation of the nozzle to head interference fit. This analytical evaluation shows that a gap exists between the nozzle outside diameter (OD) and RPVH shell during normal operating conditions thus meeting the NRC requirements of a "qualified" visual inspection.

Further, a review of U. S. PWR industry experience has revealed that all CRDM nozzles (35) with cracks through the pressure boundary were identified by visible leakage at the junction of the nozzle and the top of the RPVH (ref 2). This includes 20 nozzles at ONS that were identified as leaking by bare metal visual inspections. NDE of the 35 nozzles confirmed the presence of cracks through the pressure boundary. No cracks through the pressure boundary were found during volumetric examination of seventy-six (76) nozzles at ONS which had no visible signs of leakage. Therefore, qualified visual inspection has identified all known industry CRDM nozzles that contained through-wall pressure boundary cracks.

The ability to visually detect a nozzle leak is related to the interference fit between the CRDM nozzle and RPVH shell. The 35 nozzles previously described as leaking all had interference fits of 0.002" or less. A review of ONS-2 fabrication records shows three nozzles with an interference fit greater than 0.002". Sixty-six (66) nozzles have shell interferences that were determined to be in the range of a 0.0018" gap to a 0.0020" diametral interference fit, placing the nozzles in the range that have exhibited signs of leakage, if cracks exist. Three nozzles (Nos. 42, 47, and 59) possess interference fits of 0.0030", 0.0025", and 0.0025" respectively. Finite element analysis of the ONS-2 CRDM nozzles was performed to confirm the expected existence of a gap at normal operating conditions for all sixty-nine (69) nozzles.

<u>Circumferential Cracking</u>

Circumferential cracks represent a potential safety concern if the cracks are allowed to grow and repairs are not performed. Safety significant circumferential cracks can initiate in the nozzle above the J-groove weld. Primary water leakage from through-wall axial cracks must be present to initiate a circumferential crack. Therefore, circumferential cracks can only initiate after primary water leakage into the annulus. At ONS such leakage will continue to be found by visual inspection.

The phenomenon of circumferential cracking was initially discovered at Oconee Units 2 and 3. Examination and analysis indicate that a through wall axial crack precedes the formation of a circumferential crack. Industry and NRC analysis demonstrate that circumferential crack growth rates are such that it is highly unlikely that code limits could be reached after sixteen months of operation (Ref 12). The ASME Code criteria were used to define a critical flaw size of 270 degrees for a circumferential crack in a CRDM nozzle. This value includes a safety factor of three. The largest observed circumferential crack at Oconee was found to be 165 degrees. An NRC draft safety assessment concludes that it would require approximately 28 months for this crack to grow to a critical size (Ref 12).

Circumferential cracks will be preceded by through wall axial cracks which will result in visible leakage. Therefore, any circumferential cracks that occur during the current operating cycle will be detected by visual inspections during the upcoming refueling outage. As described above, circumferential cracks that may initiate during the upcoming operating cycle will not have sufficient time to grow to a size that could compromise code limits and thus avoiding the potential of a rod ejection. Visual inspections performed during the upcoming refueling outage provide a reasonable basis for concluding that circumferential cracks which could challenge code limits will not occur during the final operating cycle before head replacement.

Wastage

Industry experience indicates that significant boric acid induced wastage of a RPVH will not result from CRDM nozzle leaks at plants with effective visual inspection programs. Duke's visual inspection experience indicates that ONS PWSCC flaws result in CRDM nozzle leaks which result in small volumes of boric acid residue. Duke has not observed any unacceptable RPVH wastage associated with CRDM nozzle leaks. Due to slow crack growth characteristics, substantial leakage only develops after an extended time. Industry experience has shown that copious deposits of boron, stained red with corrosion products, would be present on a RPVH if subjected to leakage rates that could produce noticeable head wastage. For example a leak rate of 0.001 gpm correlates to approximately 500 in^3 of boron deposit accumulation in one fuel cycle. This volume would occupy a rectangular space which measures 10 in. x 10 in. x 5 in. and would easily be observed during routine inspections. Boron deposits observed on the ONS head are significantly less than those associated with leak rates that could produce wastage.

It is reasonable to conclude that the leakage associated with the ONS nozzle degradation is found by qualified visual inspections before it approaches the leak rates predicted to produce conditions that support wastage (Ref 9). This observation is consistent with ONS experience in that no substantial wastage of carbon steel has been observed during the investigation of any leakage on the ONS RPVHs. Further, in the course of completing repairs of the ONS nozzles by manual and automatic means, up-close and detailed inspections have been completed of the head and bore area and no evidence of significant wastage has been found.

As further additional conservative actions, Duke will implement a number of enhancements to improve the capability for identifying symptoms of RPVH degradation that could occur during operation. These enhancements include provisions focused on indications of reactor coolant system (RCS) leakage and metal wastage. These enhancements address the issues described in NRC Information Notice 2002-13 "Possible Indicators of Ongoing Reactor Pressure Vessel Head Degradation". These enhancements will include:

- Periodic entries into containment at power checking for accumulations of boric acid deposits and iron metal particles. These periodic entries will be in conformance with ALARA principles.

- Routine at-power checks of containment radiation monitor filters for the presence of boric acid and iron oxide accumulation.

- The ONS Corrective Action Program will be used to ensure that information relevant to detection of RPVH degradation is captured and promptly provided to the engineer responsible for the condition of the RCS and to Oconee senior management.

- In addition, indication of RCS and RPVH leakage and degradation can be obtained while performing routine and non-routine activities such as:

    - Pump and valve maintenance.
    - Inspection and Maintenance of Reactor Building Cooling Unit cooling coils and ductwork.
    - Results of Reactor Building sump chemistry analyses for boric acid.
    - Results of RCS leakage and Reactor Building sump inflow determinations.

Oconee is investigating other enhancements that may be useful in confirming the integrity of the RCS and RPVH for the upcoming operating cycle.

Risk Assessment for One Cycle Operation Prior to Head Replacement (Unit 2)

Framatome ANP has performed an ONS-2 specific estimation of core damage frequency (CDF) that accounts for undetected CRDM nozzle cracks and potential rod ejection that would result from a circumferential crack above the J-groove weld (ref 13). The ONS-2 risk evaluation addresses the estimated 1.3 years duration of the next fuel cycle. The next cycle of operation is projected to go from the start of cycle 20 (scheduled November 2002) through the end of cycle 20 (scheduled March 2004), when RPVH replacement is scheduled.

The ONS-2 risk evaluation is based upon the methodology of the B&W Owners Group generic risk assessment, the ONS-3 specific risk evaluation, and refinements in the methodology made as the results of several recent technical meetings with the NRC.

The risk evaluation uses the above inspection plan and information as bases, i.e. relying on a qualified bare metal visual inspection, blade probe inspection of the CRDM nozzles, and a conditional core damage frequency value of 3.5E-3 for a medium break LOCA. The risk assessment calculates a core damage frequency for the next ONS-2 operating cycle of 4.93E-7/ reactor-year which is well below the Regulatory Guide 1.174 threshold value for risk significance. These results show that PWSCC of the CRDM nozzles is not risk significant for ONS-2 during the final cycle prior to RPVH replacement.

Nuclear Fuel Damage Assessment

As an additional measure to support the other deterministic and risk analysis work discussed in previous sections of this response, Duke has performed an analysis to evaluate the potential impacts to the Oconee reactor cores should a nozzle fail and be ejected from the reactor vessel. The damage experienced following the failure of a CRDM nozzle has been reviewed to evaluate the potential for this event to cause fuel rod damage or failures. The rod ejection accident (REA) analysis has been performed for the current operating cycle and for the operating cycle prior to RPVH replacement. These analyses have been performed in accordance with the NRC-approved methodology of topical report DPC-NE-3005-PA.[1]

The analyses include simulations of the beginning-of-cycle (BOC), four reactor coolant pump (RCP) initial condition rod ejection for all of the cycles. A representative fuel cycle simulation was performed at BOC three RCP initial conditions and another at end-of-cycle (EOC) four RCP initial conditions, to demonstrate that the BOC four RCP case is the most limiting. The BOC three RCP case and the EOC four RCP cases yielded more margin to departure from nucleate boiling ratio (DNBR) than the BOC four RCP case. For the Hot Zero Power (HZP) core conditions the cycle-specific ejected rod worth is insufficient to achieve prompt criticality. Consequently, the rapid power excursion shown in DPC-NE-3005-PA for the HZP case does not occur and the resultant transient is easily bounded by the four RCP case.

The peak core power for the BOC four RCP case analyzed for ONS Unit 2, Cycle 20 is 112% of full power. This is in contrast to a similar case in DPC-NE-3005-PA; which yielded a maximum core power of 140% full power. The results of this conservative analysis of the rod ejection accident indicate that no fuel rod failures due to DNB or any other reason would be anticipated for the cycles in question. This is in contrast to the highly conservative 40.6% fuel pin census results shown in Table 14-4 of DPC-NE-3005-PA or the ~50% fuel failures assumed in the offsite dose calculations. The calculations of peak pressure and peak fuel enthalpy were not performed in these analyses as neither limit was violated in the current UFSAR Chapter 15 analyses. These relatively benign results are due to the small ejected rod worth values for the Oconee core designs of concern.

[1] Oconee UFSAR Chapter 15 Transient Analysis Methodology

Risk associated with maintenance activities are managed and documented for Oconee systems, structures and components as required by 10 CFR Part 50.65 and Duke's policies and procedures.[2] This includes compliance with paragraph (a) (4) to assess and manage the increase in risk that may result from proposed maintenance activities. Work activities are performed to provide the level of plant equipment reliability necessary for safety, and are carefully managed to achieve a balance between the benefits and potential impacts on safety, reliability and availability. Risk assessments are performed to manage the increased risk that may result from proposed work activities. Assessment of proposed work activities determines the effect of maintenance on the availability of high safety significant plant systems that have been modeled in the ORAM-SENTINEL risk assessment tool. When the proposed maintenance renders these systems unavailable, the work is coded as causing unavailability of the systems. The plant configurations that occur during maintenance are then evaluated using the ORAM-SENTINEL risk assessment software tool.

Six Concerns from NRC Bulletin 2002-02

Concern 1: Circumferential cracking of CRDM nozzles was identified by the presence of relatively small amounts of boric acid deposits. This finding increases the need for more effective visual and non-visual NDE inspection methods to detect the presence of degradation in CRDM nozzles before nozzle integrity is compromised.

Response: Since the initial discovery of circumferential cracks above the J-groove weld in 2001, visual inspection techniques and approaches have been dramatically improved. A heightened sense of awareness exists for the range in size and appearance of visual indications that must be further investigated. Non-visual techniques similarly have and continue to evolve to more effectively examine the penetration tube and associated welds for evidence of cracks. Nothing in the recent events at Davis-Besse has altered the fundamental inspection capability requirements previously established to identify the presence of PWSCC and subsequent associated wastage. The effectiveness of these inspection techniques continues to be evaluated and improved.

The MRP and EPRI have published a paper titled "Visual Examination for Leakage of PWR Reactor Head Penetrations" (Ref 1) that will be used in training personnel performing the visual inspection along with lessons learned from the previous five Oconee inspections. The ONS-2 bare metal visual inspection of the RPVH will be performed and documented in accordance with written procedures and acceptance criteria. Inspections are conducted by an experienced component engineer and a QA inspector qualified to perform VT-2 inspections. When evidence of leaks is found, the visual acceptance standards and supplemental actions meet the intent of ASME Section

[2] 10 CFR Part 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, or the Maintenance Rule.

XI with the realization that systems will not be at normal operating temperature and pressure.

In order for OD circumferential cracks above the J-groove weld to initiate and grow, a leak path must first be established to the CRDM annulus region from the inner wetted surface of the RPVH. If primary water does not leak to the annulus, the environment does not exist to cause circumferential OD cracking. Axial cracks in the CRDM nozzles or cracks in J-groove welds must first initiate and grow through wall. Oconee experience has shown that through wall axial cracks will result in observable leakage at the base of the penetration on the outer surface of the vessel. Industry experience with Alloy 600 steam generator drain pipes at Shearon Harris (1988) and pressurizer instrument nozzles at Nogent 1 and Cattenom 2 (1989) were all roll expanded but still developed leaks during operation (Ref 2). Plant specific RPVH to CRDM nozzle gap analyses have been performed for ONS-2, with nozzle initial interference fits up to 0.0030". These analyses have confirmed the presence of a head to nozzle gap sufficient to provide a physical leak path for all ONS-2 nozzles under normal operating pressure and temperature conditions.

The ONS-2 head was designed to have a nozzle to shell interference fit ranging between 0.5 and 1.5 mils. A review of fabrication records show eight nozzles with diametral interference fits greater than 1.5 mils. Five nozzles were in the range from 1.5 to 2.0 mils and three from 2.0 to 3.0 mils. A gap analysis using the fabrication dimensions determined that when the unit is at pressure and temperature, a gap exists between the nozzle and shell. This annulus creates a tightly enclosed collection point for any leakage through the nozzle base or weld metal. When the head cools, the interference between the nozzle and shell is re-established. Any boric acid deposits are forced out of the annulus, where they are easily visible at the nozzle to head junction. This understanding of the gap geometry is consistent with the extruded or string-like morphology of boron deposits observed on the Oconee heads.

The probability of detecting small CRDM leaks by visual inspection alone is high. Visual inspections of the reactor coolant system pressure boundary have proven to be an effective method for identifying leakage from PWSCC cracks in Alloy 600 base metal and Alloy 82/182 weld metal prior to the leak becoming a significant safety issue. Specifically, visual inspections throughout the industry have detected leaks in RPVH CRDM nozzles, RPVH thermocouple nozzles, pressurizer heater sleeves, pressurizer instrument nozzles, hot leg instrument nozzles, steam generator drain lines, a RPV hot leg nozzle weld, a power operated relief valve (PORV) safe end and a pressurizer manway diaphragm plate (Ref 3). Past volumetric inspections of leaking CRDM nozzles at ONS demonstrate that when a leak path through the pressure boundary material existed, boric acid deposits were visible on top of the head. In contrast, non-leaking nozzles examined to determine extent of condition exhibited no evidence of the presence of a leak path. This observation demonstrates that nozzles degraded to the extent that a leak path exists into the annulus will be detectable by the boron deposits at the nozzle to RPVH junction.

In addition to the bare metal visual inspection, Duke will perform a blade probe UT inspection to identify PWSCC within the CRDM nozzle material and to look for evidence of leakage through the shrink fit region of the annulus ("leak path" inspection). The ability of the blade probe to detect both axial and circumferential indications has been demonstrated recently using EPRI blind mockups. The "leak path" inspection provides additional assurance that leakage into the annulus has not occurred. Its ability to identify leakage has been confirmed by Framatome ANP field UT inspection results from leaking nozzles.

In summary, Oconee has responded to the need to detect small amounts of leakage by increased visual inspection sensitivity, increased inspection area, and improved inspection capabilities. Small amounts of leakage can be detected visually and it has been shown that timely detection by visual inspection will ensure the structural integrity of the RPVH penetrations with respect to circumferential cracking. The ONS-2 RPVH is clean and relatively free of pre-existing boric acid deposits. The clean head facilitates a comprehensive 100% bare metal visual 360 degrees around each nozzle. A clean head also insures that small boric acid deposits are readily detected as demonstrated by past experience. The blade probe UT inspections will ensure that the penetration tube material is free of unacceptable indications and provide assurance that leakage has not occurred.

Concern 2: Cracking of 82/182 weld metal has been identified in CRDM nozzle J-groove welds for the first time and can precede cracking of the base metal. This finding raises concerns because examination of weld metal material is more difficult than base metal.

Response: Through wall cracks in the J-groove weld pose a similar risk as nozzle base metal cracks because the cracks leak into the annulus and can initiate circumferential cracks in the nozzle above the J-groove weld. Leakage into the annulus from either through wall cracks in the nozzle or weld metal are equally detectable by visual inspection. Although higher crack growth rates have been observed in weld metal, the industry model of time-to-leakage includes plants that have had weld metal cracking as well as base metal cracking. The visual inspection performed on ONS-2 will ensure that significant degradation would not result in either rod ejection or substantial wastage prior to head replacement during the next refueling outage.

As additional conservative actions, Duke will conduct a volumetric inspection of the sixty-five nozzles using blade probe ultrasonic technology that is delivered by an automated inspection tool from under the RPVH. (The four nozzles repaired during ONS-2 EOC-18 will be excluded from blade probe inspection.) Blade probe UT inspection will be conducted by qualified personnel using equipment and procedures, with acceptance criteria, demonstrated on EPRI blind mockup test blocks witnessed by the NRC in August 2002. Blade probe UT utilizes time of flight, tip diffraction techniques (TOFD), optimized for circumferentially oriented cracks but has demonstrated ability to detect axial cracks. Blade probe scanning is effective axially to 11.25 inches which is sufficient to examine most nozzles from below the weld starting at the bottom edge of the nozzle to the top of the head. There may be some nozzles where inspection scans will not reach the top of the head. However, it is expected that the scan will cover the nozzle below the weld, the weld area and the shrink fit region. In addition to TOFD techniques, blade probe techniques will be utilized to interrogate the interference fit for detection of potential leak path or wastage. Leak path technology will be used to insure leakage through the weld has not occurred. For those cases where leak path techniques are indeterminate, further evaluation will be conducted and documented which may include PT examination of the weld metal surface.

Bare metal visual inspections and the blade probe UT provide assurance that leakage into the annulus has not occurred and no significant indications exist in the nozzle. Knowing the condition of the nozzle at the beginning of the cycle, coupled with crack growth rates in the nozzle material, makes the inspection of the weld unnecessary to prevent rod ejection by precluding the existence of a circumferential crack above the J-groove weld at the beginning of the operating cycle.

Concern 3: Through-wall circumferential cracking from the outside diameter of the CRDM nozzle has been identified for the first time. This raises concerns about the potential for failure of CRDM nozzles and control rod ejection, causing a LOCA.

Response:

Circumferential cracks in the OD of the nozzle above the J-groove weld require RCS leakage into the annulus. Past experience at ONS has demonstrated that qualified bare metal visual inspection of RPVH will detect leakage into the annulus. In addition, ONS-2 is performing volumetric inspection of the CRDM nozzles to identify any axial or circumferential cracks in the nozzles as well as looking for leak paths and wastage above the J-groove weld. All nozzles with unacceptable inspection results will be repaired prior to the unit being returned to service.

UT examination of twenty leaking nozzles at ONS showed crack like indications within the nozzle indicative of PWSCC. It is recognized that a PWSCC leak path to the annulus can be contained strictly within the weld. If a leak were to occur during startup, fracture mechanics crack growth analyses performed by the industry and NRC consultants show that it would take several cycles for a circumferential crack to grow to a size large enough to cause failure of the CRDM nozzle and rod ejection.

A plant specific risk assessment for ONS-2 was performed by Framatome ANP. This assessment assumed a qualified bare meal inspection and blade probe UT shows the probability of rod ejection is 1.41E-4/reactor year and CDF is 4.93E-7/reactor year which is well below the threshold established in Regulatory Guide 1.147.

Therefore Duke believes the inspection plan for ONS-2 that utilizes a qualified visual inspection, supplemented by volumetric blade probe inspection and an ONS specific risk assessment provides sufficient assurance that the issue of circumferential cracking is being appropriately managed.

Concern 4: The environment in the CRDM housing/RPVH annulus will likely be more aggressive after any through-wall leakage because potentially highly concentrated borated primary water may become oxygenated. This raises concerns about the technical basis for current crack growth rate models.

Response: The EPRI MRP panel of international experts on SCC (including representatives from ANL/NRC Research) gave extensive consideration to the likely environment in the annulus between a leaking CRDM nozzle and the RPVH prior to the Davis-Besse incident and subsequently revisited this issue (Ref 5). When revisited, the relevant arguments remain valid as long as leak rates are less than 1 liter/hr or 0.004 gpm, which appears to be consistent with plant experience. The conclusions of the expert panel include:

1. An oxygenated crevice environment is highly unlikely because:

   - Back diffusion of oxygen is too low compared to counterflow of escaping steam (two independent assessments based on molecular diffusion models were examined).
   - Oxygen consumption by the metal walls would further reduce its concentration.
   - Presence of hydrogen from leaking water and diffusion through the upper head results in a reducing environment.
   - Even if the concentration of hydrogen was depleted by local boiling, coupling between low alloy steel and Alloy 600 would keep the electrochemical potential low.
   - Corrosion potential will be close to the Ni/NiO equilibrium, resulting in PWSCC susceptibility similar to normal primary water.

2. The most likely crevice environments are either hydrogenated steam or PWR primary water within normal specifications and both would result in similar, i.e. non-accelerated, susceptibility of the Alloy 600 penetration material to PWSCC.

3. If the boiling interface happens close to the topside of the J-groove weld, a low probability occurrence, a concentration of PWR primary water solutes, lithium hydroxide and boric acid can, in principle, occur. The concern here would be the accelerating effect of elevated pH on PWSCC; but calculations and experiments show that any changes are expected to be small, in part because of the buffering effects of the precipitates.

Higher leakage rates into the annulus have not been analyzed by the expert panel to determine the chemistry for this condition. However, higher leakage rates would be expected to reduce the potential for any concentrated products in the annulus and result in the annulus chemistry being that of the primary water.

The above considerations show there is no basis for assuming that any post-leakage, crevice environment in the CRDM housing/RPVH annulus would be significantly more aggressive with regard to PWSCC of the Alloy 600 penetration material than normal PWR primary water, irrespective of the assumed leakage rate and/or annulus geometry. Further, the current industry model (Ref 5), includes a factor of two on predicted crack growth rate to cover residual uncertainty in the chemistry of the annulus environment.

Concern 5: The presence of boron deposits or residue on the RPVH, due to leakage from mechanical joints, could mask pressure boundary leakage. This raises concerns that a through-wall crack may go undetected for years.

Response: The ONS-2 inspection plan recognizes the potential for masking of the RPVH material or leaking CRDM nozzles. Any location on the ONS-2 head that may be obstructed by deposits will be cleaned, inspected and evaluated for any identified RPVH wastage. Any nozzle that is masked will be inspected volumetrically for PWSCC. The recent experience at another PWR plant clearly demonstrates that effective visual inspection for leakage from CRDM nozzles and welds requires unobstructed inspection access. The head surface must be free of pre-existing deposits. Accumulations of debris and boric acid from other sources that can interfere with the visual inspection must be removed.

The ONS-2 head was cleaned during the last refueling outage to optimize the effectiveness of the last visual inspection prior to head replacement. Any deposits will be removed and the masked area will be examined for wastage. Should volumetric inspection not identify an indication on a masked nozzle, the weld surface will be PT tested for disposition of the nozzle. This graded approach is consistent with the inspection and evaluation protocol that Duke has committed to in NRC Bulletin 2001-01 and is consistent with the program Duke has used in the last five outages at Oconee.

Concern 6: The causative conditions surrounding the degradation of the RPVH at Davis-Besse have not been definitively determined. The staff is unaware of any data applicable to the geometries of interest that support accurate predictions of corrosion mechanisms and rates.

Response: The causes of the Davis-Besse degradation are sufficiently well known to avoid similar significant wastage. The root cause evaluation performed by the utility (Ref 7) clearly identifies the root cause as PWSCC of CRDM nozzles followed by boric acid corrosion. The large extent of degradation has been attributed to failure of the utility to address evidence that had been accumulating over a five year period (Figure 26 of Ref 7).

Oconee and other plants have been provided guidance for reactor vessel top of head visual inspections to ensure that conditions such as that found at Davis-Besse are avoided. Visual inspection guidelines have been provided (Ref 1), and a workshop was conducted to thoroughly review industry experience, regulatory requirements, leakage detection, and analytical work performed to understand the causes of high wastage rates (Ref 8).

The ability to detect leakage prior to the risk of structural failure is illustrated by Figure 26 of the Davis-Besse root cause analysis report (Ref 7). There was visual evidence of boric acid deposits on the vessel head for five years prior to the degradation being detected. Therefore, while the exact timing of the event progression at Davis-Besse cannot be definitively established, the probable durations can be predicted with sufficient certainty to conclude that a visual inspection regimen such as practiced at ONS insure continued structural integrity of the RCS pressure boundary.

The observed boron deposits at ONS have been significantly smaller than that observed at Davis Besse. The first nozzle determined to be leaking at ONS was first detected during a routine planned end-of-cycle RPVH inspection by the observation of a boron deposit of approximately ½ cubic inch. By contrast, the boric acid deposit observed on the Davis Besse head in 1996 was reported to have blocked the visual inspection of four nozzles which would have masked an area of several square feet.

Field observations associated with the ONS heads indicate that leakage detected in a reasonable time does not result in substantial wastage of the carbon steel head. As noted previously in this response, Duke has found and repaired 20 nozzles that had evidence of visual leakage. However, only minor surface corrosion has been noted on only one of these nozzles.

Should RCS leakage occur at ONS-2, the operational leak monitoring programs are designed to detect the leakage prior to escalating to a substantial leakage event. Pressure boundary leaks from the Oconee RCS are monitored in several ways. The Operations group performs proceduralized RCS leakage calculations on a daily basis. Typically this is done by running a RCS leakage calculation program on the plant computer for a specific interval of time. The daily leakage program includes mass calculations for the RCS, Pressurizer, Letdown Storage Tank and Quench Tank. Least Squares best fit of water inventories are calculated in order to determine the RCS leakage rate. With a tight RCS, leakage values typically average between 0.05 and 0.10 gpm. The accuracy of the RCS leakage calculation program has been demonstrated through testing to be on the order of 0.05 gpm when run for specific intervals. Oconee Technical Specifications allow for an unidentified leakage rate limit of 1 gpm. However, the Oconee threshold for further investigation is typically 0.2 gpm. The leakage calculation procedure requires a second leakage calculation be performed if the leakage rate is greater than 0.2 gpm. Monitoring and trending of 5 day and 30 day average RCS leakage rates are performed by the RCS system engineer.

In addition, RCS leakage is monitored by trending other instrumentation associated with the Reactor Building. Reactor Building Normal Sump (RBNS) rates are trended daily. An increase in the sump collection rate may indicate the presence of an RCS leak. Frequent chemical analyses of the RBNS inventory may also indicate an RCS pressure boundary leak if the boron concentrations begin to increase. Reactor Building radiation monitors are trended weekly. Increasing radioactivity in the Reactor Building may also point to a RCS pressure boundary leak. Pressurizer relief valve tail pipe temperatures are trended weekly. High tail pipe temperatures with increasing Quench Tank temperatures usually indicate a leaking safety or relief valve. These measures enhance the ability to detect and discriminate leakage sources such as a leaking safety or relief valve from unknown sources such as leakage via RPVH penetrations.

NRC Regulations in 10 CFR 50.55a states that ASME Class 1 components (which include VHP nozzles) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Various portions of the ASME Code address reactor coolant pressure boundary. For example, Table IWA-2500-1 of Section XI of the ASME Code provides examination requirements for RPVH pressure retaining components and references IWB-3522 for acceptance standards. IWB-3522.1(c), (d), and (e) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residue on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as "the through-wall leakage that penetrates the pressure retaining membrane." Even though the NRC is currently questioning the inspection requirements in the ASME Code, it is clear that the ASME Code does not permit continued operation with through-wall degradation of the RPVH. Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit continued operation with through-wall degradation of the reactor pressure vessel head penetration nozzles.

Compliance

Oconee is in compliance with 10 CFR 50.55a and the criteria of IWB-3522.1 (c), (d), and (e). The ONS RPVH Inspection plan is a stepped approach utilizing a 100% bare metal visual to inspect a full 360 degrees around each of the 69 CRDM nozzles. Any evidence of leakage, head wastage or any nozzle(s) that may have masking deposits of boron or other matter around the nozzle will be noted. Deposits capable of masking wastage will be removed and the previously masked area visually inspected for wastage. Additional visual inspections of the area inside the service structure above the insulation such as inspections for flange leakage may be used to aid in the final disposition of a nozzle as leaking. The bare metal visual inspection of the RPVH will be performed and documented in accordance with written procedures and acceptance criteria. Inspections are conducted by an experienced component engineer and a QA inspector qualified to perform VT-2 inspections. The visual acceptance standards and supplemental actions required when evidence of leaks is found, meet the intent of ASME Section XI with the realization that the system will not be at normal operating temperature and pressure at the time of inspection.

Nozzles showing signs of leakage or any nozzle(s) that may have a masking deposit of boron or other matter around the nozzle will be subjected to volumetric inspections. The UT inspections will be performed by qualified personnel using procedures and equipment that were demonstrated on EPRI blind mockup test blocks that were witnessed by the NRC in August 2002. Should UT inspection fail to identify an indication on a masked nozzle, a dye penetrant test will be performed to detect possible cracking in the weld. Also, as previously described, Duke will volumetrically inspect the remaining un-repaired ONS-2 nozzles.

During previous Oconee outages, qualified bare metal visual inspections performed under the insulation have detected leakage, with detection being a result of identifiable boron deposits that had accumulated on the top of the head. Oconee performed appropriate NDE of each of these leakage events to determine the source of the leakage/deposit and then performed repairs in accordance with rules stipulated by the ASME Code and NRC approved Code alternatives. Deposits were removed and the masked area was inspected for wastage.

For through-wall leakage identified by visual examinations in accordance with ASME Code, acceptance standards for the identified degradation are provided in IWB-3142. In accordance with these requirements, methods are in place to determine the acceptability of degraded components including supplemental exams, corrective measures, or repairs, analytical evaluation and replacement.

As required by IWB-3142, all identified degraded and leaking components will be repaired prior to returning the unit to service. Repair of minor shallow ID indications may not be required if evaluation according to ASME Section XI analytical flaw evaluation rules determines acceptability for continued service.

Additionally, Oconee performs visual inspection of accessible and exposed surfaces during system pressure testing as part of their In-Service Inspection Program required by 10 CFR 50.55a. The visual examination may be conducted by looking for evidence of potential leakage. The acceptance standard for the examination is found in IWA-5250, "Corrective Measures." This subsection requires repair or replacement if a leak is identified as well as assessment of damage, if any, from corrosion of steel components by boric acid deposits.

The effectiveness of the qualified bare-metal visual inspections, the NDE performed and to be performed, and engineering analysis provide reasonable assurance that compliance with code margins and acceptance criteria will continue to be maintained for the operating period.

Criterion V (Instructions, Procedures, and Drawings) of Appendix B to 10 CFR Part 50 states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual, volumetric, and surface examinations of the reactor coolant pressure boundary are activities that should be documented in accordance with these requirements.

Compliance
Activities associated with the RPVH are performed in accordance with the Duke QA Program. Procedures which address activities associated with QA Condition 1 structures, systems and components are subjected to a well-defined and established preparation, review, and approval process as defined in the Duke QA Program. This QA Program meets Criterion V - Instructions, Procedures, and Drawings.

Visual inspection procedures will use acceptance standards and require actions which meet the intent of ASME Section XI and are consistent with the requirements for VT-2 inspections with the realization that these inspections will not be performed at normal operating pressure and temperature. For volumetric and surface examinations the acceptance standards will be to NRC guidance (Ref 14).

Criterion IX (Control of Special Processes) of Appendix B to 10 CFR PART 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of the reactor coolant pressure boundary for the degradation observed at Davis-Besse, special requirements for visual examination and/or ultrasonic testing would generally require the use of visual and ultrasonic testing methods. Such methods are ones that a plant-specific analysis has demonstrated would result in reliable detection of degradation prior to the loss of specified reactor coolant pressure boundary integrity and margins of safety. The analysis would have to consider, for example, the as-built configuration of the system and capability to reliably detect and accurately characterize flaws or degradation, and contributing factors such as access to the inspection area, the presence of insulation, preexisting deposits, and other factors that could interfere with the detection of degradation.

Compliance
Activities for characterizing and repairing pressure vessel head CRDM nozzle defects are performed in accordance with the Duke QA Program which has been reviewed and approved by the NRC. The Duke QA Program, in general, maintains procedures for the control of a number of special processes including welding, heat treating, NDE, and

cleaning. The program requires that approved, written procedures, qualified in accordance with applicable codes and standards, be utilized when it affects the performance of the station's QA Condition 1 structures, systems and components. These procedures provide for documented evidence of acceptable accomplishment of these special processes using qualified procedures, equipment and personnel as may be required by ASME Section XI for reactor coolant pressure boundary components.

Personnel performing such activities must be qualified in accordance with applicable codes and standards. Adequate documentation of personnel qualifications is required. NDE examination personnel are certified to required codes and standards.

As part of the requirements for a "qualified visual", nozzle specific evaluations have been completed using original as-built head bore and nozzle dimensions to demonstrate that at normal plant operating conditions, a positive gap will exist such that through-wall leakage evidence would be visible on the RPVH. Further, any UT or other NDE examinations performed will have been subjected to sufficient demonstration testing to substantiate the capability of the examination method. Both rotating probe UT and blade probe UT were demonstrated to the industry and the NRC using the EPRI blind mockup test blocks in August 2002 by Framatome ANP in Lynchburg Virginia.

Criterion XVI (Corrective Action) of Appendix B to 10 CFR PART 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For cracking of VHP nozzles or material wastage of the RPVH, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future cracking or material wastage. These actions could include proactive inspections, repair of leaking VHP nozzles, and valid acceptance by analytical evaluation for degraded VHP nozzles where through-wall leakage may not emanate.

Compliance

Activities associated with the RPVH are performed in accordance with the Duke QA Program. Pursuant to this program, station personnel are responsible for the implementation of the QA Program as it pertains to the performance of their activities. Specific to this responsibility is the requirement for informing responsible supervisory personnel and/or for taking appropriate corrective action whenever any deficiency in the implementation of the requirements of the program is determined. Procedures require that conditions adverse to quality be corrected. In the case of significant conditions adverse to quality, the procedures assure that the cause of the condition is determined and action be taken to preclude repetition.

Performance and verification personnel are to:

a) Identify conditions that are adverse to quality.

b) Suggest, recommend, or provide solutions to the problems as appropriate.

c) Verify resolution of the issue.

Additionally, performance and verification personnel are to ensure that reworked, repaired, and replacement items are inspected and tested in accordance with the original inspection and test requirements or specified alternatives.

In the event of the failure of QA Condition 1 components (such as degradation of the RPVH) the cause of the failure is evaluated and appropriate corrective action taken. Items of the same type are evaluated to determine whether or not they can be expected to continue to function in an appropriate manner. The evaluation is documented in accordance with applicable procedures. This corrective action program meets the requirements of Criterion XVI - Corrective Action.

Specifically, each occurrence of RPVH leakage at ONS has been rigorously and thoroughly investigated to determine the cause of the leakage condition. This investigation provides Duke with the knowledge and understanding of why these leaks have occurred and also provides Duke the ability to characterize the condition of each RPVH. The leaks found at ONS are due to PWSCC. Inspections planned to be performed during the ONS-2 RFO are a result of the Corrective Action Program.
In addition Duke has concluded that the best way to prevent further leaks is to replace each RPVH with a new RPVH that incorporates enhanced design features to better resist the PWSCC phenomena.

Technical Specifications - The current limiting condition of operation (LCO) for ONS, TS 3.4.13, requires that RCS operational LEAKAGE be limited to no pressure boundary LEAKAGE;

- 1 gpm unidentified LEAKAGE;
- 10 gpm identified LEAKAGE;
- 300 gallon per day total primary to secondary LEAKAGE through all steam generators (SGs)
- and 150 gallon per day primary to secondary leakage through any one SG. These limits are applicable in operational modes 1 through 4.

Compliance

The inspection and maintenance programs in combination with the corrective action program provides reasonable assurance of reactor pressure boundary integrity and compliance with Technical Specification 3.4.13 limits.

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Sketch #1 Showing Triple Point