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To: F. Mark Reinhart, NRR
Date: 2/6/02 1:37PM
Subject: PRA perspective in Technical Assessment REport

here is my first draft input, which include three plant-specific insights. The three plants are D.C. Cook, Davis Besse, and Oconee. Please provide me with comments by no later than February 20, 2002.

CC: Steven Long

E-54

4.0 SYSTEMS AND RISK DISCUSSION

4.1 Accidents Associated with CRDM Housing Failure

Two classes of accidents are applicable to the failure of a CRDM housing: the reactivity insertion due ejection of a control rod and a loss of coolant accident (LOCA). Each is part of the design basis for a nuclear power reactor.

4.1.1 Control Rod Ejection Accident

All PWRs must analyze the ejection of a control rod drive assembly, for which each NSSS has a unique name, and the subsequent reactivity insertion. Control rod drive housing failures and leaks are evaluated based on longitudinal as well as circumferential failures. The nuclear design is such that the energy deposited in the fuel rods adjacent to the rod that is postulated to be ejected will not exceed 280 cal/g (1.17 kJ/g), the failure threshold for uranium dioxide, zirconium-clad fuel. A typical rod ejection analysis is performed considering an average core channel and a hot region. Each is also performed for the maximum allowed rod bank insertion at a given power level. Appropriate safety analysis margins are added to the ejected rod worth and hot channel factors to account for calculational uncertainties. The results are then provided for analyses at beginning of cycle - full power, beginning of cycle - zero power, end of cycle - full power, and end of cycle - zero power.

4.1.2 Loss of Coolant Accident (LOCA) and Anticipated-Transient-Without-Scram (ATWS)

All PWRs must analyze a spectrum of LOCAs ranging from full double-ended guillotine separation of the largest reactor coolant pipe to the smallest size break, temperature has been bounded. The analyses are also performed assuming the highest-worth control rod assembly fails to insert, in some cases, and that all control rods fail to insert in other cases. The results of the analyses must meet the acceptance criteria specified in 10 CFR 50.46, specifically that neither the peak cladding temperature does not exceed 2200°F (1204.4 °C) nor that there is oxidation of more than 17% of the fuel cladding.

Preliminary evaluations of representative hardware configurations indicate that the separation of a control rod drive mechanism housing does not present any new or different thermal-hydraulic phenomena. However, the circumferential CRDM nozzle cracking at ONS-2 and ONS-3 raises concerns about potential safety implication on adequacy of public protection, where the thermal-hydraulics of the event with this break size was analyzed but the risk and initiation of the crack-induced CRDM failure were not analyzed.

The staff examined various aspects of the CRDM failure phenomena, including potential collateral damages and the event mitigation options. The objective was to identify potential worst case scenario and bounding event, and to understand risk insights associated with CRDM nozzle cracking so that an appropriate level of attention could be given to the issue. The staff focused its review on the crack initiation and propagation phenomena leading to a failure of the CRDM VHP nozzle and the consequent LOCA when the circumferential through-wall cracks are propagated to a point that the reactor coolant system pressure inside the reactor vessel would eject the CRDM. The loss of coolant depends on the size of the break, and may range from a leak to a large break LOCA, depending on the design dimensions of the relevant components, collateral damages, and failure mechanisms.

If the CRDM VHP nozzle breaks and yet the control rod does not eject completely out of the vessel penetration, leaving a relatively small flow area at the vessel head, equivalent to and less than about ½ inch (1.27 cm) in diameter. This size leak could be controlled by the normal charging and makeup system, and would be unlikely to result in a core damage event.

A small break LOCA would occur if the CRDM housing is somehow partially ejected from the reactor vessel penetration or the hole is partially blocked, leaving an equivalent flow area between ½ inch to 2 inches (1.27 to 5.08 cm) in diameter (up to 3.14 in^2 (20.25 cm^2) cross section). This break size would be least likely break, unless the broken CRDM is partially pushed out of the vessel penetration, lodged in the vessel penetration, or debris inside of the vessel partially blocks the opening. The size of such break would not be large enough to depressurize the system before core damage occurs, nor remove sufficient energy to cool RCS. Therefore, for a successful mitigation of this size LOCA it would need other means of removing energy and the RCS must be depressurized if low pressure injection is needed.

A medium break LOCA size is generally considered to be 2 to 6 inches (5.08 to 15.24 cm) in diameter (up to 28.3 in^2 or 182.58 cm^2 in cross section). This size break would be the most likely break once the separated upper part of the CRDM begins upward motion under the large RCS pressure. Even if the circumferentially cracked nozzle is ejected from the vessel penetration but the control rod drive shaft somehow lodged and stuck in the vessel penetration, the flow area would be sufficiently large enough to be considered as medium size LOCA. This size of breaks was considered as a part of design basis accident scenario and evaluated accordingly for potential fuel failures under the 10 CFR 50, Appendix K, as opposed to the containment performance for large break LOCA.

Large break LOCAs occur with break sizes 6 inches (15.24 cm) in diameter or larger. This scenario would not be likely since it would require more than two CRDM ejections simultaneously. Large breaks remove the decay heat through the breaks, and only inventory makeup is of important.

In summary, the most likelihood limiting accident sequence would be the medium size LOCA as the upper CRDM is separated by the circumferential cracking, and subsequently ejected out of the vessel penetration due the hydraulic force of the reactor coolant. Once a medium size LOCA occurs, the injection for inventory makeup would be from RWST (or BWST for B&W plants) or upper head accumulators initially. However, sooner or later, the plant operator must switch from the injection mode to the recirculation mode, drawing water from the containment sump as the makeup water tank is depleted. This switch over is plant-specific, and may require either manual operation or automatic switchover. The CE plants are normally automated for the swapping operation.

The common cause failures of pumps and valves, and potential operator failures of timely switch over to recirculation are the major risk contributors to failures of the event mitigation, and consequently, the conditional core damage probability (CCDP) is relatively high for such failures. This insight is based upon plant-specific Individual Plant Examination (IPE) under the GL 88-20. Data from these studies indicate the CCDP, given that a medium size LOCA has occurred, falls in the range of 1E-3 to 1E-2.

Core damage contribution from the anticipated-transient-without-scram (ATWS)-type events or the reactivity perturbation is practically negligible. However, potential complications can occur due to collateral damage to other CRDMs, or blockage of the emergency core cooling recirculation sump, or delay in switch over to sump recirculation, and projectiles from the CRDM failure inside containment. Such complications may increase the risk associated with a CRDM failure.

4.2 Risk Perspectives and Adequate Protection

The Commission Policy on use of probabilistic approach was published in Federal Register Vol. 60, No. 158, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities: Final Policy Statement" on August 16, 1995. It affirms that probabilistic risk assessment (PRA) methods can be used to derive valuable insights, perspective, and general conclusions as a result of an integrated examination of facility design, response to initiating events, expected interactions among structures, systems, and components, and with its operating staff. It endorses the use of the risk insights in conjunction with regulatory requirements and the defense-in-depth philosophy in the decision making process.

The Regulatory Information Summary (RIS) 01-002, "Guidance of Risk-Informed Decision making in License Amendment Reviews" dated January 18, 2001, further clarifies some of the ambiguity of NRC's policy on the implementation and use of PRA. The RIS addresses safety principles of risk-informed decision making, and provides guidance to identify a "special circumstance" in which compliance with Commission regulations does not implicitly address a safety issue for adequate protection of the public. It provides a process for the staff to consider whether a "special circumstance" exists which may rebut the presumption that compliance with the regulations provides adequate protection of public health and safety. Although developed as a tool for staff reviews of license amendment requests, the process in the RIS is appropriate for other regulatory decision making purposes because it addresses the fundamental requirement for operation of a nuclear reactor: there is reasonable assurance of adequate protection for the public health and safety.

A "special circumstance" is present if compliance with current regulatory requirements does not provide appropriate means to detect and protection against degradation of plant hardware, and deficiency of plants operation, and thus, assure the structural integrity and protect against a severe accident. Failure of the regulations to require adequate protection that could lead to a failure of structural integrity, and consequently a severe accident, constitutes a risk factor addressed by the Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

In the RG 1.174 a "special circumstance" exists and acceptable if the following five principle criteria are met:
(1). It meets current regulations. This first criterion addresses the licence conditions and the requirements of 10CFR50.55a.

(2). It is consistent with "defense-in-depth philosophy," The second criterion must be satisfied because, compliance with the regulations may not be adequate to prevent the failure of the reactor coolant pressure boundary, one of the three barriers to release of radioactive materials from the reactor core.

(3). It maintains sufficient safety margin, For CRDM cracking. The compliance with the ASME Code, Section XI, inservice inspection requirements fails to satisfy the third principle of maintaining safety margins since it cannot be assured that pressure boundary leakage would be detected prior to a gross failure of a vessel head penetration nozzle.

(4). It results in only a small increase in core damage frequency. The fourth principle addresses application of the numerical guidance in the RG 1.174, which further provides for an acceptable level of change in risk for a given change in licensing condition, consistent with the Commission safety goal. The extension of temporary plant operations under the adverse conditions, such as existence of CRDM cracking but with an assurance of safety margin and acceptable remedial actions, may be evaluated based on the temporary risk addressed in the RG 1.182 under the 10CFR50.65 maintenance rule.

Regulatory Guide 1.182, "Assessing and Managing Risk before Maintenance Activities at Nuclear Power Plants" May, 2000, was aimed to monitor the overall effectiveness of the licensee maintenance programs, and established methods that are acceptable to NRC staff. As an attachment to the RG 1.182, Section 11, "Assessment of Risk Resulting from Performance of Maintenance Activities," dated February 11, 2000, of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," specifies temporary risk ceiling of 10^{-3} per reactor year for temporary configuration-specific CDF, and for which the NRC neither endorses nor disapproves. However, Table in the section 11.3.7.2 established risk management action thresholds based on quantitative considerations for incremental core damage probability (ICDP) and incremental large early release probability (ILERP) for the configuration-specific risk. The ICDP or ILERP can be evaluated by integrating the CDF or LERP over the time duration of the temporary configuration.

(5). Finally, the basis for the risk estimate is monitored using performance measurement strategies. the fifth principle is satisfied if the basis for any analysis that shows risk levels below Regulatory Guide 1.174 numerical guidelines are based on assumptions that can be verified and are capable of detecting the form of degradation being modeled.

Summarizing the above, it would be appropriate to use risk insights for decision making process under the fourth and fifth principles. The regulatory guides cited here recommend to use mean values for the numerical guidelines.

4.3 Probabilistic Risk Assessment

4.3.1 CDF and LERF

The frequency of the LOCA initiation depends on the likelihood of a crack and its circumferential propagation, followed by a CRDM VHP nozzle failure. Thus, an increase in plant risk would depends on the frequency of the CRDM failure and the likelihood of the recovery failure from the event.

The quantitative understanding of a crack initiation requires appropriate evaluation of fracture phenomena based on the probabilistic fracture mechanics. The mitigation of the event or the CCDP, given the initiation of the nozzle failure, would require understanding of optional success paths and their failure probability of the mitigation systems as well as operator actions to prevent continuous progression of the LOCA event, which would ultimately lead to core damage.

RG 1.174 provides numerical guidelines for acceptable level of increase in core damage frequency (CDF) as a result of plant adverse event, as given in Figures 3 and 4 for CDF and large early release frequency (LERF) respectively. As a sample application of the acceptance guidelines in RG 1.174, for a medium break LOCA with 1E-3 CCDP, the initiating event frequency of 1E-2/yr or higher could result in an unacceptable increase in CDF if plants baseline CDF lies between 1E-5/yr and 1E-4/yr.

The point estimates of crack initiation and propagation risk numbers at this time would be very unreliable due to lack of knowledge and absence of verifiable inspections whether or not cracks existed and their severity, if it did exist. In fact, additional information on the environment, material properties, residual stress and evaluation of cracking are needed as input to reliable fracture mechanics models. However, some risk informed insights are estimated utilizing operating experience and probabilistic fracture mechanics work performed by ANL. As noted in **Section.....**circumferential crack lengths as large as 165° have been found in one of the high susceptibility plants....[insert Jack's other comments]. The effort to estimate the initiating event frequency will continue as additional data become available in the future. The initiating event frequency of the CRDM cracking, along with proper consideration of the defense-in-depth and appropriate safety margin, will help guide the staff to take appropriate courses of actions on the issue in order to maintain public health and safety. Such actions should include risk management in terms of prediction, prevention, control, and mitigation of the CRDM nozzle cracking issue. From a short-term risk standpoint, compensatory measures may be prudent for the high susceptibility plants.

4.3.2 Probabilistic Fracture Mechanics and Initiating Event Frequency

One of the major objectives of PFM evaluation is to demonstrates that the PFM model represents VHP nozzle cracking and failure phenomena accurately, and an appropriate analysis of the model parameters in the crack initiation model is employed. However, high degree of uncertainty in the parameters for estimating the probability of crack occurrence and size of circumferential cracks raises more questions than answers. Such uncertainty prevents the staff from concluding that the probability of gross nozzle failure is sufficiently small, and the resulting conditional core damage probability would satisfy the numerical guidance in criterion 4 of the RG 1.174.

There are more than one approache employed by both industry and NRC to evaluate the flaw initiation and the rate of a crack propagation, either using the inspection findings or an appropriate PFM model. Use of the verified cracks and other known data tends to give relatively larger flaw rates as compared with the results based on statistical analysis employing Weibull distribution and parametric evaluation. The materials characteristics and exposure temperature and time plays a major role on the flaws. However, the high degree of uncertainty leads staff to believe that frequent qualified inspections could provide more realistic and accurate picture of the flaw initiation and distribution of the crack sizes.

The industry and NRC appear to agree on the crack propagation model in general, although there are subtle differences in details and applications. Some of such disagreement include the initial shape, location, size, and distribution of the cracks as well as associated stress profiles, impact of residual stress, and resulting crack propagation.

The bounding study of crack growth rate appears to be the analyses based on the heat 69, the worst case for a material characteristics, using linear distribution of the initial crack sizes and a 95/50 crack growth curve. At present, utility supported by Industry in general chose to use 50 percentile median curve, which tends to give approximately 2.5 times slow growth rate. Staff believes that truth lies somewhere between two but closer to heat 69 and 95 percentile curve.

Summarizing the above, staff believes that the predicted leak rates may be represented by Weibull leak initiation model with upper 95% projection with1.5 Shape Factor unless otherwise the Shape parameters are scaled or modified with appropriate justification. NRC internal study by a contractor indicated that it would be prudent to use 95/50 likelihood projection for a crack growth rate as opposed to median 50/50. Where, 95/50 represents 95% population (data) with 50% confidence level. Again, NRC staff believes that the crack growth rate should be based on the worst heat, heat 69, with the upper 95 percentile (95/50) bounding projection instead of 50/50 median or 75% estimates.

There are several unresolved issues beside the issue discussed above, such as use of linear or non-linear crack size distribution, critical circumferential crack size for failure, initial flaw verses crack growth rate, and RCS temperature. The stress profiles may dictate the crack growth rates, if given same other conditions. Failure to conduct inspections of the reactor VHP nozzles in a manner that is sufficient to detect the extent of degradation caused by a mechanism known to be degrading other similar plants in that portion of the vessel and prior to a significant reduction in safety margin could increase risk significantly as well as the associated uncertainty. Based on a licensee response submitals, Staff performed a sensitivity study, and the result indicated that a good qualified inspection may impact on the initiation frequency as much as five times.

4.3.3 Event Mitigation and CCDP

A typical success criteria to mitigate a medium break LOCA is dictated by timely injection of cooling water to replace the RCS loss through the break, providing core cooling and coverage. Because of the limited amount of water stored in a borated water storage tank and continuous need to remove decay heat beyond the capacity of the stored water, the core cooling water source has to be swapped over to a new makeup water source (such as sump) before the stored water runs out. Most PWR plants may have been designed to perform the switchover automatically (particularly CE design) or manually from the control room. However, they often requires equipment lineup and/or valve opening/closure, and reset breaker interlocks from outside of the control room.

For the LOCA due to the CRDM failure, the major risk contributors, therefore, are human failure(s) of switchover from injection-to-recirculation phase in timely manner, followed by failure(s) of high and low pressure injection phase.

Reliability and failure rates of hardware may not be easy to improve permanently within a relatively short time period. But the availability of the hardware and reduction of human errors can be minimized by reducing on-line maintenance and surveillance activities with additional or dedicated operator using well written procedures.

According to the IPE database, the CCDPs for PWR plants are in the range of 10^{-2} to 10^{-3} for a medium LOCA with a few outliers. The CCDP values from IPE database are tabulated in Tables 4.1, 4.2 and 4.3 for B&W, CE and Westinghouse designs respectively.

4.3.4 Temporary Configuration Risk and Shutdown Delay

When a plant safety-related equipment is out-of-service temporarily, the plant risk would be increased temporarily until the downed equipment is restored. Similarly, when a plant with a known deficiency extends its operation for a short period of time, the incremental risk by the deficiency during the extended operating period may be considered as an additional temporary risk. However, one of the most difficult issue is how to deal with the degraded situation and assessment of potential or projected risk. As a conservative approach, it would be prudent to treat the degraded function or component as a failure if the degraded situation can not be identified by conventional inspection techniques and a complete failure is suspected or imminent prior to scheduled outage.

A CDF is normally based on an annualized core damage probability (CDP), and is calculated using a mean, time-averaged value from a long term cumulative core damage probability, which includes both emerging and scheduled maintenance risk. It may also include other factors that may influence plant configurations and associated operational risk. In general, other than plant modifications, the on-line and emerging maintenance works, or even surveillance tests, are considered as temporary risk contributors and expressed as incremental CDP (ICDP). Therefore, the temporary risk contributors are dependent on plant configurations and varies with time, and would return to a baseline risk as soon as the temporary risk contributors are removed.

The temporary increase of core damage probability has been addressed in the EPRI "PSA Applications Guide," August 1995, which was the source reference document for the guidelines in RG 1.174 and RG 1.182, and was the driving factor under the a(4) requirements of the maintenance rule 10CFR50.56. The RG 1.182, "Assessing and Managing Risk before Maintenance Activities at Nuclear Power Plants", May, 2000, was intended for managing short-duration or transitional risk, and its attachment 2 provided numerical guidelines for ICDP and ILERP for temporary risk increases.

It restricted any increase in cumulative increase of core damage probability greater than 10^{-6} for temporary changes, again using mean values. For CRDM cracking, the cumulative core damage probability due to the cracks can be obtained based on the increase of CDF as a result of the CRDM cracking until the corrective actions are taken and the deficiency is removed. Such increase in CCDP may be obtained by integrating time-dependent initiating event frequency from the probabilistic fracture mechanics.

If the increase in CDF value for CRDM-induced MLOCA by 10^{-5} per year, it is considered non-risk-significant temporary increase if the plant operates with the event configuration for less than 5.6 weeks since the ICDP during this period would be less than 10^{-6}.

Furthermore, it would be prudent to take certain compensatory measures to reduce a plant CCDP temporarily. Some of the measures may include the following steps:

1. Reduction of the vessel temperature by reducing RCS temperature. The temperature reduction may slow down the crack initiation and propagation rates.

2. Minimizing on-line activities to reduce temporary risk configurations (i.e., reducing on-line maintenance), and

3. Reduction of CCDP by increasing a reliability of plant systems with multiple trains and by reducing potential human errors during plant operation (i.e., training or additional operators with clearly written instructions). This step may even reduce the CCDP affecting other accident sequences, and thus, would reduce the baseline risk).

4.3.5 Integrated Decision Making and Uncertainty

It is important to characterize the impact of uncertainty in the risk analysis, and to recognize them for the decisionmaking process. In fact, the decision should not be made by the numerical values of the PRA only. They are an input to an overall picture of the risk change although they plays an important role to put the change into an appropriate context in big picture, as a part of the five principles discussed in section 4.2.

For CRDM cracking issues, there are concerns related to the nature of the epistemic uncertainty, such as the confidence associated with the risk analysis and consideration of details for the decisionmaking process. where, such details may include 1). the parametric issues in the modeling initiating events such as Poisson or PFM model, 2). the modeling uncertainty associated with human performance, common cause failures, CRDM failure phenomena, Poisson and Binomial models for running and on-demand failures of components, and 3). completeness of modeling due to unanalyzed events, and shutdown operations as examples.

Certain specific guidelines were given in the RG 1.174, such as use of mean risk values or mean distribution for numerical assessment. Because of such specific approaches recommended in RG 1.174, it can be shown that undesirable artificial margin or unexpected and unqualified factors can be embedded and thus maked in the risk values. This is not consistent with the intent of the RG when the numerical guidelines are applied to the quantitative risk values derived from a median or even higher percentile value or distribution are used. Furthermore, certain compensatory measures may be taken during plant operation but not reflected in the risk model. In such cases, the numerical guidelines in the RG should be qualified and the impact of such measures should be characterized qualitatively. Also, such arguments should be considered in the decisionmaking process.

One conflicting aspect of using RG 1.174 for the CRDM risk assessment process is that the crack growth rate was evaluated based on median or 75 percentile distribution by industry, and staff recommended to use more bounding 95%. Now, this would apparently introduce additional uncertainty for using the threshold guideline given in the RG 1.174

6.0 DESCRIPTION OF BULLETIN 2001-01

The discoveries of cracked and leaking Alloy 600 VHP nozzles at four PWRs raised concerns about the potential safety implications and the prevalence of cracking in VHP nozzles in PWRs. Therefore, on August 3, 2001, the NRC issued NRC Bulletin 2001-01 "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" to all holders of operating licenses for PWRs. The purpose of the bulletin was to request information related to the structural integrity of the reactor pressure vessel head penetration nozzles at PWR facilities. Specifically, the NRC requested information on the extent of VHP nozzle leakage and cracking found to date, inspections and repairs undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements.

6.1 Summary of Bulletin

The recent identification of circumferential cracking in CRDM nozzles at ONS2 and ONS3, along with axial cracking in the J-groove welds at these two units and at ONS1 and ANO1, has resulted in the staff reassessing its conclusion in GL 97-01 that cracking of VHP nozzles is not an immediate safety concern. Specifically, the findings indicate that circumferential cracks outside of the J-groove welds can occur, in contrast to an earlier conclusion that the cracks would be predominantly axial in orientation. The findings indicate that cracking of the J-groove weld metal can precede cracking of the base metal. These findings raise questions regarding the industry approach, developed in generic responses to GL 97-01, that utilizes PWSCC susceptibility modeling based ~~on the~~ base metal conditions and do not consider those of the weld metal. Further, the findings at ONS2 and ONS3 highlight the possible existence of
a more aggressive environment in the CRDM housing annulus following through-wall leakage; potentially highly concentrated borated primary water could become oxygenated in this annulus and possibly cause increased propensity for the initiation of cracking and higher crack growth rates.

These occurrences reinforce the importance of conducting effective examinations of the RPV upper head area (e.g., visual under-the-insulation examinations of the penetrations for evidence of borated water leakage, or volumetric examinations of the CRDM nozzles), and using appropriate NDE methods (such as PT, UT, and eddy-current testing) to adequately characterize cracks. Because of plant-specific design characteristics, there is no uniform way to perform effective visual examinations of the RPV head at PWR facilities. However, one aspect of conducting effective visual examinations that is common to all PWR plants is the need to successfully distinguish boric acid deposits originating with VHP nozzle cracking from deposits that are attributable to other sources.

The Electric Power Research Institute Report TP-1001491, Part 2, "PWR Materials

Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44), Part 2: Reactor Vessel Top Head Penetrations," uses an assessment of the relative susceptibility of each PWR to OD-initiated or weld PWSCC based on the operating time and temperature of the penetrations. Based upon this simplified model, each PWR plant was ranked by the MRP according to the operating time in EFPY required for the plant to reach an effective time-at-temperature equivalent to ONS3 at the time the above-weld circumferential cracks were identified in early 2001. From the results of the susceptibility rankin g model, the population of PWR plants can be divided into several subpopulations with similar characteristics:

- those plants which have demonstrated the existence of PWSCC in their VHP nozzles (through the detection of boric acid deposits) and for which cracking can be expected to recur and affect additional VHPs;
- those plants which can be considered as having a high susceptibility to PWSCC based upon a susceptibility ranking of less than 5 EFPYs from the ONS3 condition;
- those plants which can be considered as having a moderate susceptibility to PWSCC based upon a susceptibility ranking of more than 5 EFPYs but less than 30 EFPYs from the ONS3 condition; and
- the balance of plants which can be considered as having low susceptibility based upon a susceptibility ranking of more than 30 EFPYs from the ONS3 condition.

Although the industry susceptibility ranking model has limitations, such as large uncertainties and no predictive capability, the model does provide a starting point for assessing the potential for VHP nozzle cracking in PWR plants. The following paragraphs characterize the suggested gradation of inspection effort for the subpopulations of plants noted above.

For the subpopulation of plants considered to have a low susceptibility to PWSCC, based upon a susceptibility ranking of more than 30 EFPY from the ONS3 condition, the anticipated low likelihood of PWSCC degradation at these facilities indicates that enhanced examination beyond the current requirements is not necessary at the present time because there is a low likelihood that the enhanced examination would provide additional evidence of the propensity for PWSCC in VHP nozzles.

For the subpopulation of plants considered to have a moderate susceptibility to PWSCC based upon a susceptibility ranking of more than 5 EFPY but less than 30 EFPY from the ONS3 condition, an effective visual examination, at a minimum, of 100% of the VHP nozzles that is capable of detecting and discriminating small amounts of boric acid deposits from VHP nozzle leaks, such as were identified at ONS2 and ONS3, may be sufficient to provide reasonable confidence that PWSCC degradation would be identified prior to posing an undue risk. This effective visual examination should not be compromised by the presence of insulation, existing deposits on the RPV head, or other factors that could interfere with the detection of leakage.

For the subpopulation of plants considered to have a high susceptibility to PWSCC based upon a susceptibility ranking of less than 5 EFPY from the ONS3 condition, the possibility of VHP nozzle cracking at one of these facilities indicates the need to use a qualified visual examination of 100% of the VHP nozzles. This qualified visual examination should be able to reliably detect and accurately characterize leakage from cracking in VHP nozzles considering two characteristics. One characteristic is a plant-specific demonstration that any VHP nozzle exhibiting through-wall cracking will provide sufficient leakage to the RPV head surface (based on the as-built configuration of the VHPs). Secondly, similar to the effective visual examination for moderate susceptibility plants, the effectiveness of the qualified visual examination should not be compromised by the presence of insulation, existing deposits on the RPV head, or other factors that could interfere with the detection of leakage. Absent the use of a qualified visual examination, a qualified volumetric examination of 100% of the VHP nozzles (with a demonstrated capability to reliably detect cracking on the OD of a VHP nozzle) may be appropriate to provide evidence of the structural integrity of the VHP nozzles.

For the subpopulation of plants which have already identified the existence of PWSCC in the CRDM nozzles (for example, through the detection of boric acid deposits), there is a sufficient likelihood that the cracking of VHP nozzles will continue to occur as the facilities continue to operate. Therefore, a qualified volumetric examination of 100% of the VHP nozzles (with a demonstrated capability to reliably detect cracking on the OD of the VHP nozzle) may be appropriate to provide evidence of the structural integrity of the VHP nozzles.

6.2 Summary of Regulatory Issues

6.2.1 Applicable Regulatory Requirements

Several provisions of the NRC regulations and plant operating licenses (Technical Specifications) pertain to the issue of VHP nozzle cracking. The general design criteria (GDC) for nuclear power plants (Appendix A to 10 CFR Part 50), or, as appropriate, similar requirements in the licensing basis for a reactor facility, the requirements of 10 CFR 50.55a, and the quality assurance criteria of Appendix B to 10 CFR Part 50 provide the bases and requirements for NRC staff assessment of the potential for and consequences of VHP nozzle cracking.

The applicable GDC include GDC 14, GDC 31, and GDC 32. GDC 14 specifies that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized; the presence of cracked and leaking VHP nozzles is not consistent with this GDC. GDC 32 specifies that components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leaktight integrity; inspection practices that do not permit reliable detection of VHP nozzle cracking are not consistent with this GDC.

NRC regulations at 10 CFR 50.55a state that ASME Class 1 components (which include VHP nozzles) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Table IWA-2500-1 of Section XI of the ASME Code provides examination requirements for VHP nozzles and references IWB-3522 for acceptance standards. IWB-3522.1(c) and (d) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residues 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." Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall cracking of VHP nozzles.

For through-wall leakage identified by visual examinations in accordance with the ASME Code, acceptance standards for the identified degradation are provided in IWB-3142. Specifically, supplemental examination (by surface or volumetric examination), corrective measures or repairs, analytical evaluation, and replacement provide methods for determining the acceptability of degraded components. Criterion IX 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 VHP nozzles, special requirements for visual examination would generally require the use of a qualified visual examination method. Such a method is one that a plant-specific analysis has demonstrated will result in sufficient leakage to the RPV head surface for a through-wall crack in a VHP nozzle, and that the resultant leakage provides a detectable deposit on the RPV head. The analysis would have to consider, for example, the as-built configuration of the VHPs and the capability to reliably detect and accurately characterize the source of the leakage, considering the presence of insulation, preexisting deposits on the RPV head, and other factors that could interfere with the detection of leakage. Similarly, special requirements for volumetric examination would generally require the use of a qualified volumetric examination method, for example, one that has a demonstrated capability to reliably detect cracking on the OD of the VHP nozzle above the J-groove weld.

Criterion V of Appendix B to 10 CFR Part 50 states that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplish~~ed in~~ accordance with these instructions, procedures, or drawings. Criterion V further states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual and volumetric examinations of VHP nozzles are activities that should be documented in accordance with these requirements.

Criterion XVI 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, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future cracking. These actions could include proactive inspections and repair of degraded VHP nozzles.

Plant technical specifications pertain to the issue of VHP nozzle cracking insofar as they require no through-wall reactor coolant system leakage.

6.2.2 Deficiencies in Current Regulations

The NRC regulation at 10 CFR 50.55a requires licensees to perform system pressure testing, VT-2, of the reactor pressure boundary in accordance with the inservice inspection requirements of Section XI of the ASME Boiler and Pressure Vessel Code. The examination and frequency requirements are contained in Table IWB-2500-1, Category B-P and the acceptance standards are contained in IWB-3522. The requirements for the system pressure tests are contained in IWA-5000.

The provisions of IWA-5242 state that a VT-2 visual examination may be conducted without the removal of insulation by examining the accessible and exposed surfaces and joints of the insulation. However, operating experience indicates the need for bare metal examinations since small leakage can result from a large crack as shown in the ONS3. This is an area where the Code requirements are insufficient in identifying flaws in VHPs.

The Code provisions for disposition of flaws is adequate. However, there is a lack of information about crack size and growth rate. In a letter to NEI dated September 24, 2001, the staff described methods for flaw characterization, acceptance, and crack growth rate which should provide reasonable assurance of structural and leak tight integrity of VHPs in light of the new operating experience. As quoted from the letter, these methods are described below.

Flaw Characterization

Flaws must be characterized by both their length and depth. There is currently insufficient data available to assume an aspect ratio if only the flaw length has been determined.

**The proximity rules of ASME Code Section XI for considering flaws as separate may be used (Figure IWA 3400-1).*

**When a flaw is detected, its projections in both the axial and circumferential directions shall be determined.*

**Flaws that are equal to or greater than 45-degrees from the vertical centerline of the CRDM nozzle, or those that are within plus or minus 10-degrees of the angle (if less than 45-degrees) that the plane of the partial-penetration attachment weld (J-groove weld) ~~makes~~ with the vertical centerline of the CRDM nozzle, are considered to be circumferential flaws.*

**The location of the flaw relative to the top and bottom of the J-groove weld shall be determined since the potential exists for development of a leak path if a flaw progresses up the nozzle past this weld. The flaw acceptance criteria are as specified below depending on whether the flaw is in the pressure boundary or in the portion of the nozzle below the J-groove weld.*

Flaw Acceptance Criteria

CRDM Nozzle Pressure Boundary

The CRDM nozzle pressure boundary includes the J-groove weld and the portion of the nozzle projecting above the weld. While the CRDM nozzle is an integral part of the reactor vessel, no flaw evaluation rules exist for nonferritic vessels or parts thereof in Section XI. Therefore, the rules for austenitic piping shall be applied with the following exceptions:

**The allowable flaw standards for austenitic piping in Section XI, IWB-3514.3 may be applied for inside diameter (ID) initiated axial flaws only.*

**The rules of IWB-3640 shall apply and the margins maintained after crack growth is evaluated for the period of service until the next inspection. The maximum flaw depth* allowed by IWB-3640 is 75-percent of the nozzle thickness (refer to crack growth rate below).

*All outside diameter (OD) initiated flaws, regardless of orientation (axial or circumferential), shall be repaired.

*All ID-initiated circumferentially oriented flaws shall be repaired.

*Any flaw detected in the J-groove weld, its heat affected zone (or adjacent base material) must be repaired. Alternatives to Code required repairs will be considered for approval if justified.

CRDM Nozzle Below the J-Groove Weld

*Axially oriented flaws (either ID- or OD-initiated) are acceptable regardless of depth as long as their upper extremity does not reach the bottom of the weld during the period of service until the next inspection.

*Circumferential flaws (either ID- or OD-initiated) are acceptable provided that crack growth is evaluated for the period of service until the next inspection. In no case shall the projected end of cycle circumferential flaw length exceed 75-percent of the nozzle circumference.

*Intersecting axial and circumferential flaws shall be removed or repaired because of the greater propensity to develop into loose parts. Note: while flaws below the J-groove weld have no structural significance, loose parts must be avoided.

Crack Growth Rate

CRDM Nozzle Pressure Boundary

*Crack growth to be used for axial ID initiated flaws shall be determined from Crack Growth and Microstructural Characterization of Alloy 600 Vessel Head Penetration Materials, by Bamford, W. H., and Foster, J. P., EPRI, Palo Alto, CA:1997. TR-109136 (Proprietary).

*There is currently no accepted crack growth rate for the Alloy 182 J-groove weld material.

CRDM Nozzle Below the J-Groove Weld

*The crack growth rate to be used for the flaws in this region of the nozzle, shall be the same as ~~that~~ used for ID initiated axial flaws within the CRDM nozzle pressure boundary.

7.0 RESPONSES FROM HIGH SUSCEPTIBILITY PLANTS

High susceptibility plants are those defined as being within 5 EFPY of ONS3 or those which have previously experienced either leakage from or cracking in VHP nozzles. Twelve plants are in this category. They include ONS1, ONS2, ONS3, ANO1, Donald C. Cook Unit 2, North Anna Units 1 and 2, H. B. Robinson Unit 2, Davis Besse Nuclear Power Station, Surry Units 1 and 2, and Three Mile Island Unit 1. For plants within 5 EFPY of ONS3, the bulletin requested that licensees provide information on future inspections and the basis for concluding that these inspections will assure that regulatory requirements are met. For plants which have previously experienced leakage or cracking in VHP nozzles, the bulletin requested that in addition to information on future inspection plans, licensee describe the extent of VHP nozzle leakage and cracking and describe the corrective actions taken in response to the identified cracking. The staff's evaluation of the licensees' responses are as follows.

7.1 Plants That Have Identified Cracking

7.1.1 Oconee Units 1, 2, And 3

7.1.1.1 Summary of Licensee Response

By letter dated August 28, 2001, Duke Energy Corporation (the licensee) submitted the Bulletin 2001-01 response for ONS1, ONS2, and ONS3.

<u>Description of VHP Nozzles and Insulation</u>
The Duke response referenced the information provided in the MRP-48 report regarding the VHP nozzles in the Oconee units. All three units have 69 CRDM nozzles, with an OD of 101.6 mm (4.001 in.) and a wall thickness of 15.7 mm (0.618 in.). In addition, ONS1 has eight thermocouple nozzles, with an OD of 26.2 mm (1.030 in). and a wall thickness of 5.3 mm (0.208 in.).

The Oconee units have Babcock and Wilcox (B&W) designed vessels with metal reflective insulation that is located on a horizontal plane above the head. The insulation is located such that the lowest clearance for inspection of the nozzles is at the top of the he~~ad, w~~hich has an approximate 51 mm (2 inch) between the upper-most nozzles and the insulation. The reactor vessel head service structure was modified to provide nine, 12-inch diameter access ports which permit access to the top of the RPV head for inspection purposes.

<u>Findings and Activities for Unit 1</u>

On November 25, 2000, evidence of reactor coolant system (RCS) leakage was identified at ONS1, as described in licensee event report (LER) 269/2000-006, Revision 1, dated March 1, 2001. As summarized in the Duke response to the bulletin, CRDM nozzle 21 and five of the eight thermocouple nozzles were identified with leakage deposits.

CRDM nozzle 21 had a single crack that originated in the J-groove weld and grew through the weld and nozzle base metal, penetrating into the annulus region to create a leak path. The crack was completely ground out of the J-groove weld and nozzle material, and the nozzle was restored to its original configuration with the shielded metal arc welding process using Alloy 690 weld material (Alloy 152). A protective Alloy 690 weld pad was applied to the repairs to protect and isolate any remaining original Alloy 600 from the reactor water environment.

For the thermocouple nozzles in ONS1, eddy current (EC) examinations and ultrasonic tests (UT) showed that all eight nozzles contained deep crack-like indications that were predominantly axial in orientation and located adjacent to (extending both above and below) the J-groove weld elevation. Repairs to the thermocouple nozzles involved removing the nozzles from service by machining out the existing nozzles and installing Alloy 690 plugs into the remaining penetration. As with the CRDM nozzle repair, a protective Alloy 690 weld pad was applied to the repairs to protect and isolate any remaining original Alloy 600 from the reactor water environment.

Metallurgical samples were taken from the CRDM nozzle 21 weld and from several thermocouple nozzles to determine the cause of the observed cracking. In additional, seven additional randomly selected nozzles were examined using eddy current (EC) testing, and a total of eighteen nozzles were also inspected using a 0° ultrasonic testing (UT) scan.

Findings and Activities for Unit 2

On April 28, 2001, a visual examination of the ONS2 RPV head identified boric acid deposits around four CRDM nozzles (numbers 4, 6, 18, and 30), as described in LER 270/2001-002, Revision 0, dated June 25, 2001. PT examinations of these four nozzle identified multiple rejectable indications on each of the four nozzles. The LER concluded that the leak paths for these four nozzles was axial cracks that initiated near the toe of the fillet weld and propagated axially along the OD interface of the nozzle and the weld.

EC examinations of the four leaking nozzles identified clusters of multiple axial indications that were located both above and below the J-groove weld. No ID initiated circumferential indications were found.

UT examinations on the four leaking nozzles identified 36 axial OD indications, and one circumferential OD crack above the weld on nozzle 18. The circumferential crack on nozzle 18 had a reported length of 1.25 inches and a depth of 0.07 inches.

Repairs of the four leaking nozzles were accomplished using a remote semi-automated repair method. For these repairs, the existing nozzle was severed at a location above the J-groove weld and then removed from the RPV head after separation from the J-groove weld. A semi-automated welding tool utilizing the gas tungsten arc welding (GTAW) process was used to install a new Alloy 690 weld material (Alloy 152) between the shortened nozzle and the inside bore of the RPV head base material.

Findings and Activities for Unit 3

On February 18, 2001, a visual examination of the ONS3 RPV head identified boric acid deposits around nine CRDM nozzles (numbers 3, 7, 11, 23, 28, 34, 50, 56, and 63), as described in LER 287/2001-001, Revision 0, dated April 18, 2001.

Penetrant test (PT) examinations of the nine suspected leaking nozzles covered an area 3 inches in diameter from the nozzle, including the J-groove weld surface, the fillet weld cap and part of the vessel head cladding, and extended 1 inch down the outside diameter of the nozzle from the weld to nozzle interface. For all nine nozzles, the PT examination revealed multiple rejectable indications. Post-repair PT examinations of nozzles 50 and 56 identified through-wall circumferential cracks extending approximately 165° around the nozzles.

EC examinations of the nine leaking CRDM nozzles and nine non-leaking CRDM nozzles (numbers 4, 8, 10, 14, 19, 22, 47, 64, and 65) indicated clusters of shall axial type cracks located above and below the weld. Nozzles 50 and 56 exhibited "non-typical clusters" above the weld; these clusters were later determined to be associated with through-wall circumferential cracks extending approximately 165° around the nozzles. Six of the leaking nozzles (numbers 11, 23, 28, 50, 56 and 63) had deep axial indications, and nozzles 50 and 56 had circumferential indications below the weld.

UT examinations were performed on the nine leaking CRDM nozzles and the same nine non-leaking CRDM nozzles examined with EC. The nine non-leaking nozzles did not have any crack-like axial or circumferential indications. The nine leaking nozzles had a total of 36 axial indications, nine circumferential indication below the weld and three circumferential indications above the weld. CRDM nozzle 23 was identified with two circumferential indications below the weld and one circumferential indication above the weld. The latter was discovered through a third party review of the data.

The leaking CRDM nozzles were repaired using manual repair methods, using Alloy 690 filler materials (Alloy 152). A protective Alloy 690 weld pad was applied to the repairs to protect and isolate any remaining original Alloy 600 from the reactor water environment.

Additional Inspections in Response to Identified Leakage

As described above, the following additional examinations were performed on non-leaking nozzles:

Oconee Unit 1: EC of ID of seven nozzles limited UT of ID of 17 nozzles

Oconee Unit 3: EC of ID of nine nozzles UT of ID of nine nozzles

Planned Future Inspections

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In its bulletin response, the licensee indicated that it plans to replace the RPV heads of all three Oconee units, beginning with the [redacted] refueling outage (RFO) for ONS3, the [redacted] RFO for ONS1 and the [redacted] RFO for ONS2. The latter two head replacements will be concurrent with steam generator replacements at these units. As described at a public meeting on September 7, 2001, the new head will use Alloy 690 CRDM nozzles along with Alloy 152 weld metal. In addition, the design will minimize the volume of weld metal on the nozzles and will include stress conditioning of the outer surface of the welds.

For the RFOs prior to the head replacements (fall 2001 for ONS3, spring 2002 for ONS1 and [redacted] for ONS2), qualified visual examinations, as described in the bulletin, will be performed for all three units. As described at a public meeting on September 7, 2001, analyses by the licensee conclude that a leakage pathway from the J-groove weld area to the outer surface of the RPV head exists for all but one of the nozzles for these three units. This single nozzle, in Unit 1, will be examined using a volumetric technique at the next RFO for this unit.

If evidence of leakage is found, the licensee committed to perform additional examinations, including PT, EC and UT, on the leaking CRDM nozzles to characterize the nature and extent of cracking. The licensee indicated that additional inspections of other nozzles will be based on the nature of the observed cracking, the extent and severity of the cracking, the occupational exposure rates, the availability of NDE equipment and a trained and qualified workforce.

Basis for Compliance with Regulatory Requirements

The licensee concludes that volumetric examinations by December 31, 2001, of its units are not necessary to provide assurance that the Oconee units will not experience significant leakage, rapidly propagating failure or gross rupture. This conclusion is based on the extensive efforts undertaken by the licensee on all three units within the last 12 months, the technical evaluations and other activities conducted to characterize and understand the situation at Oconee. The licensee includes an Oconee-specific risk assessment, described above, to buttress its conclusion that volumetric examination by December 31, 2001, is not necessary.

Risk Assessment

In response to Bulletin 2001-01 and to supplement deterministic analysis of CRDM nozzle for three Oconee units, the risk analysis was performed to evaluate the plant risk for continuous operation with potentially undetected CRDM nozzle cracks during the period prior to vessel head replacement. The plant-specific risk assessment was performed based on the risk assessment completed by Framatome-ANP for the B&W Owner's group. The analysis considered that the limiting event due to the CRDM cracking would be a medium size LOCA upon failure of the nozzle and subsequent CRDM ejection, leading to core damage. The licensee employed following assumptions and methods:

a. Based on the flaws identifiedd by inspection at three Oconee units and ANO-1, the licensee assumed that the flaws were initiated over tha last two operating cycles (18 month cycle). ~~With 15~~ leakers at 4 plants during past 12 reactor years (1.5 year per cycle and two cycles for 4 units), the leak rate of 1.25 per reactor year was estimated for the crack initiation. Once, OD is wetted, 100% of the OD cracks are assumed with zero-time-initiation.

b. The probability of OD flaw-to-CRDM failure in one cycle was considered with a probability of 1.3×10^{-2}. Same probability number is also used for the second cycle.

c. The probability of 6% not detecting an existing leak due to human error was included. Again, 6% failure rate is used for inspection efficiency and this was an area which NRC staff was not prepared to give credits to previous refueling outage inspections.

d. The flaw-to-CRDM failure was calculated based on a Monte Carlo simulation of the fracture mechanics and one failure per 80,000 Monte Carlo simulation is used (1.3×10^{-2}). In the crack propagation, initial flaw size from 0 to 180 degrees in linear distribution was assumed, and crack growth was simulated using Peter Scott model, which is a function of stress intensity and temperature.

e. The CCDP for MLOCA is $3.5x10^{-3}$, based on Oconee PRA, Revision 2, December 1996. The initiating event frequency of MLOCA due to CRDM failure is $1.73x10^{-5}$ per reactor year.

f. The CDF contribution due to CRDM failure-induced MLOCA is $6.0x10^{-8}$ per reactor year, well below the threshold value of $1.0x10^{-5}$ per reactor year in RG 1.174.

g. The conditional population dose (CPD) with a medium size LOCA would be relatively small since the MLOCA may not challenge the containment performance. Thus, the CPD value of $1.1x10^{-4}$ person-rem is used with a resulting public health risk of $6.6x10^{-4}$ person-rem per year.

7.1.1.2 Staff Assessment

[Additional input from Allen]

The probability of CRDM crack-to-failure propagation appears to be too optimistic. However, the assumption employed for initial crack size distribution is conservative. An independent analysis by a NRC contractor indicated that median value for the crack growth rate may not represent the actual rate, considering number and sizes of crack observed in many plants, and a use of 95% value would be more prudent and bounding approach.

Based on the worst heat (heat 69) with the 95 percentile crack growth rate curve, the initiating event frequency of the MLOCA would be almost two order of magnitude lager than the value reported by the licensee. However, even if the 95% value is used the CDF increase due to the CRDM failure-induced-MLOCA would meet the threshold CDF guideline in RG 1.174. Furthermore, the RG 1.174 guidelines and threshold numbers were provided based on the mean values, and the acceptance guidelines may be higher if 95% values were used. Staff concluded that the risk assessment result provided ample margin.

7.1.2 Arkansas Nuclear One Unit 1

7.1.2.1 Summary of Licensee Response

By letter dated September 4, 2001, Entergy Operations, Inc. (the licensee) submitted the Bulletin 2001-01 response for Arkansas Nuclear One Unit 1 (ANO1).

Description of VHP Nozzles, Insulation, and Configuration

ANO1 has 69 reactor vessel head penetrations containing 68 CRDMs and one reactor vessel level instrument which is the center nozzle. ANO1 is a Babcock and Wilcox (B&W) designed vessel with metal reflective insulation that is located on a horizontal plane above the head. The lowest clearance for inspection of the nozzles is at the top of the head, which has an approximate 2-inch space between the upper-most nozzles and the insulation. The reactor vessel head service structure support contains more than 20 opening which allow inspection in the base of the skirt around the vessel head.

Detection and Repair of Leaking CRDM

In licensee event report (LER) 50-313/2001-002-00 dated May 8, 2001, the licensee described their actions associated with discovery of pressure boundary leakage. During the Spring 2001 refueling outage, the licensee saw evidence of boric acid leakage around CRDM nozzle number 56 during a visual inspection. After this initial inspection, and with the reactor vessel head still on, the licensee expanded the visual examination to all CRDM nozzles using remote video equipment. No other leakage was observed during this expanded inspection. Penetrant examinations (PT) were performed on the Alloy 600 J-groove weld-to-nozzle number 56 from beneath the reactor vessel head. The PT examinations found a crack on the outer diameter of the nozzle beneath the weld (below the reactor coolant pressure boundary) on the downhill side. The crack contained a circumferential segment at a location 0.4 inches from the weld fusion line, then the crack curved into the axial direction. The PT examinations did not detect any indications on the inside surface of the nozzle.

An ultrasonic (UT) examination confirmed that there was a leak path at CRDM number 56 that extended from an OD crack that propagated partially through-wall past the weld to the nozzle annulus. The licensee performed an embedded flaw weld repair where the circumferential portion of the flaw was removed by severing the nozzle just above its circumferential extent. The axial portion was removed in the J-groove weld and on the OD of the nozzle by grinding. The excavated cavity was built back up using an Alloy 690 compatible weld material.

Additional Inspections in Response to Identified Leakage

Framatome ANP performed an automated UT and ET examination of the CRDM nozzle after completing repair of the J-groove weld. The results confirmed that the remaining embedded flaw was unaffected by further welding activities. The licensee evaluated the need to perform additional inspections on other CRDM nozzles during the Spring 2001 outage, and concluded that additional examinations were unnecessary. The basis for this conclusion was bounding fracture mechanics and flaw growth evaluations which showed that adequate safety margin exists to ensure that no adverse structural concern would exist between refueling outages assuming significant initial flaws.

Planned Future Inspections

The licensee committed to perform a qualified visual examination of essentially 100% of the outer bare metal surface of the CRDMs during the [redacted] outage. The licensee also committed to develop contingency plans for volumetric examination if necessary. With regard to long term management of primary water stress corrosion cracking (PWSCC), the licensee has an effort in progress to develop a mitigation technique that would apply a weld overlay of corrosion resistant material to the wetted surface of the CRDM nozzle and J-groove weld using remote automated tooling. The licensee noted that the technique, once developed, could be applied as a repair or preventative action for cracking of CRDM penetrations. Although not a commitment, the goal is to begin using the technique in the [redacted] outage.

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If evidence of leakage is found, the licensee committed to perform additional examinations on the leaking CRDM nozzles to characterize the nature and extent of cracking. The licensee did not identify what the scope of expansion would be for inspection of additional CRDM nozzles.

Risk-Assessment

The licensee stated that a probabilistic fracture mechanics evaluation is in progress by the EPRI MRP that will provide an estimate of the likelihood of a pipe rupture in the CRDM penetrations.

Basis for Compliance with Regulatory Requirements

The licensee concluded that the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CRDM nozzle cracking are clearly in compliance with regulatory requirements.

7.1.2.2 Staff Assessment

The staff reviewed the ANO1 Bulletin 2001-01 response, and had the following specific comments. With regard to identification of further leakage, it should be noted that new flaw acceptance criteria have been developed by the NRC which will be forwarded to the industry to facilitate dissemination of the information. In addition, the NRC concludes that all CRDMs should be volumetrically examined upon discovery of additional leakage. Since leakage has been previously identified at ANO1, the licensee may be subject to enforcement action if further leakage is identified. The basis for this approach is that further leakage may suggest inadequate corrective action.

The licensee included a discussion on risk assessment, and stated that the NRC's assumption of an initiating event frequency of 1 for a rupture of a CRDM penetration is extremely conservative. The initiating event frequency of 1.0. may have been used by the staff as part of sensitivity calculations to evaluate the CCDP given a CRDM penetration rupture, but is not considered a staff assumption. The staff acknowledges that the risk assessment is not complete, but is in progress by the EPRI MRP.

The staff concluded that the proposed method and timing of inspection are acceptable. The staff is satisfied that the licensee's inspections during the Spring 2001 outage were sufficient to ensure that no structurally significant flaws were left in service, and the qualified visual inspection may be performed at the next scheduled refueling outage (Spring 2002).

7.1.3 Donald C. Cook Unit 2

7.1.3.1 Summary of Licensee Response

By letter dated September 4, 2001, Indiana Michigan Power Company (the licensee) submitted the Bulletin 2001-01 response for Donald C. Cook Units 1 and 2 (D.C. Cook Units 1 and 2).

Description of VHP Nozzles, Insulation, and Configuration

D.C. Cook Unit 2 has 79 reactor vessel head penetrations containing 73 CRDM nozzles, 5 thermocouple nozzle, and one head vent nozzle. D.C. Cook Unit 2 is a Westinghouse designed vessel with metal reflective insulation. The seismic support structure provides lateral stability for the CRDM housings as well as access for the interconnecting cables, and is anchored to the refueling cavity wall.

Detection and Repair of CRDM

D.C. Cook Unit 2 was in an extended outage from September 1997 to June 2000 which limited the amount of EFPY of operation. The licensee has not detected leakage from the vessel head penetrations. However, during the Fall 1994 refueling outage, an EC examination performed on 71 of the 78 vessel head penetrations showed indications in CRDM nozzle number 75. Three, closely spaced axial indications were found. The upper extent of one indication was near the J-groove weld, but the flaw was primarily below the weld. A UT examination confirmed the largest indication, however the two smaller indications did not show up separately because they were too shallow (< 1mm) or because of their proximity to the larger indication. The licensee completed a flaw evaluation which provided the justification for continued operation. The CRDM nozzle was repaired by embedding the flaw using an alternate repair method which the NRC approved by letter dated April 9, 1996. The technique partially removed the flaw and a weld overlay was applied.

Additional Inspections in Response to Identified Flaw

During the 1996 refueling outage, the five outer vessel head penetrations, including CRDM nozzle number 75, were re-inspected using the same EC examination technique used in 1994. re-inspection of CRDM nozzle number 75 showed no significant flaw growth, and no additional indications were identified in the other four outer penetrations. As mentioned above, CRDM nozzle number 75 was subsequently repaired after the re-inspection.

Planned Future Inspections

The licensee committed to perform a remote visual examination of all accessible vessel head penetrations under the reactor vessel head insulation during the next refueling outage in Fall 2001. In addition, the licensee committed to perform EC examination of the vessel head penetration base material near the susceptible weld area and the J-groove welds. Any relevant indications will be investigated using a UT technique to size and characterize their depth, length, and orientation. The licensee will re-examine the embedded flaw in CRDM nozzle number 75 using a liquid penetrant technique to verify that there are no surface indications open to the primary water environment. All detected flaws will be evaluated for acceptability using the criteria contained in the vendor's flaw data handbook which the licensee stated was under development. The handbook will contain predetermined evaluations for flaws dependent on size, location, and orientation that will permit determination of the way the flaw may be dispositioned. The licensee also stated that the scope of enhanced examinations beyond the next (Fall 2001) inspection has not been determined.

Basis for Compliance with Regulatory Requirements

The licensee concluded that the provisions described in the response to the Bulletin provide reasonable assurance that the vessel head penetration reactor coolant pressure boundary is not breached, and will assure that the applicable regulatory requirements are met.

Risk Assessment

D.C.Cook has developed an initiation scenario of the CRDM crack-induced LOCA event based on industry experience, coupled with the engineering judgement. The correlation between industry experience, engineering judgement, and risk quantification was not clearly explained in the response. The results are as following:

a. The cumulative probability of first leak was estimated as a function of the operating time, EFPY, based on statistical analysis of industry data. The cumulative probability was converted into yearly leak frequency by taking the slope of the probability-versus-time relationship (the equation presented in the attachment 1 of the Bulletin resopnse for the frequency calculation has an error in the denominator by omitting time increments). The initial through-the-wall leak rate frequency of 1.07 x10^{-5} per reactor year is reported.

b. After the first leak as defined as axial through-the-wall crack, the crack would propagate circumferentially and to the opening sizes which would lead to leak, SLOCA, and MLOCA , and may lead to core damage. Licensee employed scott crack growth model to evaluate the crack propagation, and median distribution was presented as opposed to NRC's 95/50 percentile values. However, in its risk assessmen the licensee presented risk model based on engineering judgement, which assumed that each paths leading to leak, SLOCA and MLOCA would occur 90%, 8% and 2% respectively. That would give the initiating event frequency of leak, SLOCA and MLOCA as 9.63x10^{-3} , 8.56x10^{-4} and 2.14x10^{-4} per reactor year with corresponding CCDP values of 3.31x10^{-5} , 3.31x10^{-3} and 4.52x10^{-3} respectively. The corresponding CDF contribution by the leak, SLOCA and MLOCA will be 3.19x10^{-7} , 2.83x10^{-6} , and 3.87x10^{-6} per reactor year. This value would be well within the acceptable CDF increase under the RG 1.174, although no clear explanation of the assigned split fractions for each event sequence was identified.

c. To extend the plant operation for 19 days from December 31, 2001 to January 19, 2002, the incremental CDP was given as 2.16x10^{-7}, which would clearly met 1.0x10^{-6} ICDP, the increase of CDP for the operating duration until January 19, 2002, as recommended in the RG 1.182 under the maintenance rule for temporary risk increase.

7.1.3.2 Staff Assessment

The staff reviewed the D.C. Cook Unit 2 Bulletin 2001-01 response, and had the following specific comments. With regard to identification of further leakage, it should be noted that new flaw acceptance criteria have been developed by the NRC which will be forwarded to the industry to facilitate dissemination of the information. The vendor's flaw data handbook that is currently under development should have criteria that are at least as conservative as the NRC staff's newly developed flaw acceptance criteria. In addition, the NRC concludes that all CRDMs should be volumetrically examined upon discovery of additional leakage or degraded vessel head penetrations. Since a flaw was previously identified at D.C. Cook Unit 2, the licensee may be subject to enforcement action if further cracking of vessel head penetrations and/or leakage is identified. The basis for this approach is that further cracking and leakage may suggest inadequate corrective action.

The staff concluded that the proposed method and timing of inspection are acceptable. The staff is satisfied that the licensee's proposed EC and UT techniques will be effective in identifying and characterizing any flaws in the vessel head penetrations, and the qualified visual inspection may be performed at the next scheduled refueling outage (Fall 2001).

Staff also believe that the methodology employed for the risk assessment was not conservative. The initial leak frequency for "i"th point given in the equation of the attachment 1 is not only wrong for omitting time interval in the denominator but also is not conservative for using points "i" and "i-1", instead of points "i" and "i+1". This would amount to 4% smaller result.

Second concern was an engineering judgement of assigning probability of 90%, 8% and 2% for the Leak, SLOCA and MLOCA respectively as the initial crack would propagate further. No basis or rationale of assigning the probability of each path from initial leak to core damage was explained. It appears that the probability numbers were arbitrarily assigned. For example, for a small LOCA, the vessel head opening created by the failure of the CRDM nozzle has to be either partially blocked by debris or the CRDM is partially ejected and somehow stuck-tilted in the vessel penetration leaving a hole equivalent to 2" diameter or smaller. It would be more likely that once the CRDM fails and the CRDM nozzle is separated from the vessel head, the vessel opening would be equivalent to the ID of the nozzle (2.75" ID) as minimum, or even the OD (4") assuming a clean ejection of the CRDM. In fact, it would be prudent to assume that once the CRDM nozzle failed it would be ejected out of the vessel penetration. In addition, staff did not see any clear connection between the facture mechanics of Scott crack growth rate and the engineering judgement employed for the risk assessment.

However, considering all of the ambiguity and uncertainty in the methodology, the initiating event frequency for LOCA is same order of magnitude compared with the NRC crack growth estimates, and the risk numbers appear to be within the bound of the acceptable guidelines in RG 1.174.

7.2 Plants That Have Not Identified Cracking but Are Within 5 EFPYs of ONS3

7.2.1 North Anna Unit 1 and Surry Unit 1

7.2.1.1 Summary of Licensee Response

By letter dated August 31, 2001, Virginia Electric and Power Company (the licensee) submitted the Bulletin 2001-01 response for North Anna Units 1 and 2 and Surry Units 1 and 2. Prior to the Bulletin response, the licensee met with the NRC staff on August 2, 2001 to discuss contingency plans for the repair of circumferential cracking of reactor vessel head penetration nozzles if such cracking is found during inspections at North Anna and Surry . A meeting summary dated August 7, 2001 was issued with a non-proprietary version of the meeting presentation materials. The NRC staff informed the licensee that two requests for relief from American Society of Mechanical Engineers (ASME) Code requirements would be necessary to implement these repair techniques. The first relief would allow the use of technical criteria from a later version of the ASME Code to support an embedded flaw repair process, and the second relief would allow the use of Code Case N-638 for an ambient temper bead weld repair technique. The relief requests were submitted to the NRC staff, and are currently under expedited review. The staff conducted two conference calls with the licensee on September 14 and 21, 2001 to discuss their inspection plans and results for North Anna Unit 1.

<u>Description of VHP Nozzles, Insulation, and Configuration</u>

North Anna Unit 1 and Surry Unit 1 each have 66 reactor vessel head penetrations containing 65 CRDM nozzles and one head vent nozzle. North Anna Unit 1 and Surry Unit 1 are Westinghouse designed vessels with stepped reflective stainless steel insulation. The service structure (also called the reactor vessel lifting rig) bolts directly to the upper head of the reactor vessel. A work platform on top of the service structure provides access to the upper CRDM housings.

Previous Vessel Head Penetration Inspections

The Bulletin requested a description of the VHP nozzle and reactor pressure vessel (RPV) inspections that have been performed in the past four years, and the findings. During the Spring 2000 outage, the licensee completed visual inspections of North Anna Unit 1 and Surry Unit 1 in accordance with Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The inspections were performed with the insulation on the head. No evidence of leakage was detected. Westinghouse performed what the licensee characterized as a "best effort" under the head non-destructive examination inspection at North Anna Unit 1 in February 1996 examining the two outermost rows of the CRDMs. The inner diameter (ID) of 20 of 65 CRDM nozzles was characterized by eddy current (EC). No reportable indications were found, however, the thermal sleeves in some penetrations interfered with the EC blade probe limiting the extent of the examination in those cases. The EC technique was only qualified to characterize axial ID cracks.

Planned Future Inspections

During the August 2, 2001 meeting on the proposed weld repair technique, the licensee stated that they will use EC on the inside diameter of the CRDM nozzles as well as the corresponding full J-groove welds from under the reactor vessel head for North Anna Unit 1 (contingent on timing of qualification). If the technique could not be qualified in time for the North Anna Unit 1 inspection in Fall 2001, the licensee planned to use the EC technique at Surry Unit 1. The licensee also stated that UT will be performed if any EC indications are found. In the Bulletin response, the licensee committed to perform an effective visual inspection of each of the CRD housings and the reactor head vent where they penetrate the top of the reactor vessel head for North Anna Unit 1 and Surry Unit 1 during the Fall 2001 outages. During a conference call on September 14, 2001, the staff reiterated the Bulletin's recommendation for a qualified visual inspection for high-susceptibility plants. During a conference call on September 21, 2001, the licensee informed the staff of their intention to qualify the visual inspections for North Anna Units 1 and 2 and Surry Units 1 and 2. The licensee also stated that they do not plan to do EC examinations on 100% of the CRDM nozzles in North Anna Unit 1 or Surry Unit 1 since they plan to qualify the visual inspection with a plant specific analysis. In their Bulletin response, the licensee committed to develop contingency plans for volumetric examination if necessary.

Planned Additional Inspections if Leakage is Detected

If leakage is detected, the licensee stated that it is their intention to perform supplemental inspections from under the head using EC and UT procedures to locate the source of leakage and to characterize any flaws that are found. The licensee also stated that expansion of the EC and UT inspections would be based on statistical determination of a relevant sample size, and any additional unacceptable indications would likely result in inspection of all of the housings on the reactor vessel head.

Basis for Compliance with Regulatory Requirements

The licensee concluded that the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CRDM nozzle cracking are clearly in compliance with regulatory requirements.

7.2.1.2 Staff Assessment

[Additional information will be provided by Andrea]

7.2.2 Three Mile Island Unit 1

7.2.2.1 Summary of Licensee Response

Exelon/AmerGen submitted it's response to Bulletin 2001-01 for the Three Mile Island, Unit 1 (TMI Unit 1) in a letter dated August 31, 2000. TMI Unit 1 is considered to be a high susceptibility plant, which is 4.1 EFPY from that of ONS3.

Description of VHP Nozzles, Insulation, and Configuration

The TMI Unit 1 reactor vessel head has 69 CRDM nozzles and 8 thermocouple nozzles. It is a B&W designed head with reflective horizontal insulation. The minimum clearance between the bottom of the insulation and the dome of the reactor vessel head surface is approximately 2 inches. TMI Unit 1 has eight 12 inch diameter access ports in the service structure.

Previous Inspections

Visual inspections were conducted during previous refueling outages in October 1997 and October 1999. In each case, a VT-3 inspection was performed of the reactor vessel head under the insulation. No evidence of leakage from the CRD and thermocouple nozzles was found. In addition, a VT-2 inspection was also conducted during each outage. Again, no evidence of leakage was detected.

Planned Future Inspections

Each refueling outage TMI Unit 1 will perform a qualified bare metal visual VT-3 inspection of all VHP nozzles. These inspections will be performed by certified ASME Level III inspectors trained in accordance with the EPRI Visual Training Package and specifically trained on VHP nozzle leakage experience from Oconee and ANO.

The TMI Unit 1 reactor head will be cleaned to remove existing deposits and videotaped prior to unit restart.

For any VHP nozzle that is identified and suspected of leaking, a volumetric examination (using best available technology) will be performed for flaw confirmation and characterization. If the characterizations indicate circumferential cracking above the J-groove weld, TMI Unit 1 will perform additional volumetric examinations of other readily available CRDMs (the CRDMs that are removed from the reactor vessel head to facilitate affected nozzle repair).

Basis for Compliance with Regulatory Requirements

The licensee concluded that the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CRDM nozzle cracking are clearly in compliance with regulatory requirements.

7.2.2.2 Staff Assessment

The staff reviewed the TMI Unit 1 Bulletin 2001-01 response, and had the following specific comments. TMI Unit 1 does not have an interference fit for the CRDM nozzles, so the visual inspections will automatically be qualified (i.e. the as built clearances between the CRDM nozzles and the reactor head penetrations would allow a leakage path). With regard to evaluation of identified flaws, it should be noted that new flaw acceptance criteria have been developed by the NRC which will be forwarded to the industry to facilitate dissemination of the information. In addition, the NRC concludes that all CRDMs should be volumetrically examined upon discovery of any leakage.

The licensee refers to the "EPRI Visual Training Package," the staff is not familiar with this package and is not sure if it is specifically designed for CRDMs.

During the licensee's response it is stated that "MRP 2001-050 indicates that the Oconee nozzles would have taken more than 4-5 EFPY to reach the structural margin. TMI Unit 1 is approximately 4 EFPY from reaching the Oconee Unit 3 time-at-temperature when the cracks were detected. Therefore, TMI Unit 1 is not expected to have any structurally significant flaws." The licensee mis-interpreted the meaning of the EPRI susceptibility ranking. The ranking is not meant to be predictive of the extent of flaws. It is a simplistic model meant to rank a unit's time-at-temperature relative to the time Oconee Unit 3 found cracks.

7.2.3 H. B. Robinson Unit 2

7.2.3.1 Summary of Licensee Response

The Carolina Power and Light company responded to Bulletin 2001-01 for the H.B. Robinson Steam Electric Plant, Unit No. 2 in a letter dated 4 September 2001. This plant is within 5 EFPY of ONS3.

Description of VHP Nozzles, Insulation, and Configuration

The HBRSEP Unit 2 RPV head has 69 CRDM penetrations. It is a Westinghouse design that was fabricated by Combustion Engineering. The insulation on the head is "blanket Contoured." This insulation was installed during the May 2001 outage, replacing the older metallic thermal insulation. The new insulation lies on the reactor head and is installed in two layers, with the blankets traversing the head between the VHPs.

Previous Inspections

VT-2 visual inspection was conducted during the last three outages which took place in 1998, 1999 and 2001. For the April, 2001 visual inspection, the inspectors were briefed and viewed videotapes of the previous experiences at Oconee with regards to VHP leakage. After the bare metal examination, it was concluded that no upward boron leakage pattern existed as seen within the ONS3 videotape.

Planned Future Inspections

The next reactor vessel head inspection is scheduled fo[redacted]A qualified bare metal visual inspection will be performed. The previous inspection in 2001 will be used as a baseline for the future inspection. Should modeling and analysis not be able to qualify the visual exam, a supplemental response will be provided to

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the NRC. Plans for future reactor vessel head inspections may be modified to incorporate "lessons learned" from other utilities and to assure that proposed inspection techniques will produce accurate and reliable results.

Should future visual examinations identify VHP nozzle leakage, appropriate actions will be taken in accordance with plant procedures ans ASME Code requirements to characterize the associated cracks or flaws. Inspection of additional VHPs using appropriate NDE techniques would be performed in order to establish the extent of condition.

Basis for Compliance with Regulatory Requirements

Carolina Power and Light concluded that the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CRDM nozzle cracking are clearly in compliance with regulatory requirements.

7.2.3.2 Staff Assessment

Bulletin 2001-01 states that a qualified visual examination of 100% of the VHP nozzles needs to be performed for plants within 5 EFPY of ONS3. This qualification needs to be plant specific, based upon the as built configuration of the VHPs. If a qualified visual examination can not be performed, a qualified volumetric examination of 100% of the VHP nozzles may be appropriate. During 2 conference calls in September, 2001, the licensee stated that they do not have as-built dimensions of their vessel head penetrations, rendering (in the opinion of the staff) it impossible for the licensee to demonstrate qualification of the plant's last visual examination and any future visual examinations. In addition the licensee was not able to provide sufficient clarification of their previous inspection to demonstrate that the visual examination performed would have been effective in detecting boric acid deposits at the CRDM nozzles.

The response is also vague with regard to the scope of future examinations in the case of leakage being detected during the qualified visual examination.

7.2.4 Davis-Besse Nuclear Power Station

7.2.4.1 Summary of Licensee Response

By letter dated September 4, 2001, the FirstEnergy Nuclear Operating Company (the licensee) submitted the Bulletin 2001-01 response for the Davis-Besse Nuclear Power Station (DBNPS).

Description of VHP Nozzles and Insulation

DBNPS has 69 CRDM nozzles, of which 61 are used for CRDMs, 7 are spares, and one is used for the RPV head vent piping. Each CRDM nozzle has an outside diameter (OD) of 101.6 mm (4.001 in.) and a wall thickness of 15.7 mm (0.618 in.).

DBNPS has a B&W-designed RPV with metal reflective insulation that is located on a horizontal plane above the head. The insulation is located such that the lowest clearance for inspection of the nozzles is at the top of the head, which has an approximate 51 mm (2 inch) between the upper-most nozzles and the insulation. The reactor vessel head service structure has 18 "mouse holes" or "weep holes" located around its circumference that permit access to the top of the RPV head for inspection purposes.

Past Examinations of the VHP Nozzles

DBNPS has performed two inspections of the VHP nozzles within the last four years. The last inspection was in April 2000 at RFO 12. Visual examination of the RPV heads near the CRDM nozzles indicated some accumulation of boric acid deposits. These deposits were positively attributed to five leaking CRDM flanges. No visible evidence of nozzle leakage was detected. For future reference, video documentation of the head condition was made after the head was cleaned with demineralized water.

Future Plans for VHP Nozzle Examinations

The licensee indicated plans to perform a qualified visual examination of the RPV head during the next RFO, scheduled for April 2002. This examination will use the basic requirements of ASME VT-2 inspection.

Because of significant efforts being undertaken by the MRP and the nuclear industry to better understand VHP nozzle cracking and to develop optimized inspection methods, mitigation and repair techniques, the licensee proposed to provide a final response to Bulletin Request 3.a by January 29, 2001, 60 days before the start of the next RFO for DBNPS.

The bulletin response provided a rationale for the qualification of visual examination for the DBNPS RPV head. The CRDM nozzles were designed with a diametral interference of 0.025 mm ± 0.013 mm (0.001 in. ± 0.0005 in.), with individual CRDM nozzle shafts custom ground to a diameter 0.025 mm (0.001 in.) greater than the final CRDM bore diameter. From measurements of the nozzle and penetration diameters, the interference fits for DBNPS range from a maximum of 0.053 mm (0.0021 in.) to a gap of 0.025 mm (0.001 in.).

From analyses performed by B&WOG and cited by the licensee, a nominal interference fit of 0.025 mm (0.001 in.) opens to a gap of 0.084 mm (0.0033 in.) when considering temperature and pressure dilation of the RPV head at operating conditions. Therefore, the licensee concludes that a leakage path of gap of less than 0.084 mm (0.0033 in.) will occur at operating conditions for DBNPS, effectively qualifying the DBNPS RPV head for a qualified visual examination.

Basis for Compliance with Regulatory Requirements

The licensee's bulletin response includes a variety of bases for concluding that the applicable regulatory requirements will continue to be met until the licensee performs inspections at DBNPS in April 2002. The bases for the various regulatory requirements include:

- The operating time before DBNPS

would reach an equivalent degradation time as ONS-3 is at least 3.1 EFPY.

- Flaw growth calculations by Framatome-ANP in April 2001 indicate that a through-wall flaw 180° around the nozzle would take approximately 4 years to grow another 25% (e.g., 90°) around the circumference.
- Failure of a single CRDM nozzle is bounded by both the LOCA and non-LOCA plant analyses, and simultaneous multiple CRDM nozzles will not fail [sic].
- DBNPS Emergency Operating Procedures provide adequate directions to mitigate any transient that would occur should there be a failure of a CRDM nozzle.

Risk Assessment

In its response to Bulletin 2001-01, Davis Besse evaluated the risk contribution of the CRDM failure and subsequent MLOCA. The onset of the cracks was evaluated based on Weibull distribution of leak initiation model, and further propagation of the cracks with modified Peter Scott metod and Heat 69 crack growth rate. The median values were used for numerical values of the risk contribution. However, the licensee claimed in its original response partial credits on inspections conducted in 1996(RFO 10), 1998 (RFO 11) and 2000 (RFO 12). An independent sensitivity study by NRC staff indicated that each inspection credit taken by the licensee resulted in almost 80% reduction in the CDF.

However, without giving any credits to previous three inspections, the CDF contribution with median crack growth rate would be close to the threshold values recommended in the RGs 1.174 and 1.182, which were based on mean values. If a bounding analysis employed 95% growth rate as recommended by NRC contractor and the above three inspections were not accepted, the risk numbers would fall above the threshold acceptable range of the RGs. In fairness, the threshold values in the Rgs should be also adjusted to 95% percentile, in order to use the risk numbers other than mean values.

With partial credits to RFO 10 inspection and using the bounding analysis on crack growth, the CDF increase due to the CRDM failure may lie somewhere between the above two values, median and 95%. Again, if the uncertainty and ambiguity of using mean values in the RGs are incorporated into the threshold values of the RG guidelines, the CDF increase would be a borderline value to be acceptable for continuous operation. Other hand, extension of the operation for additional one and a half month (approximately one tenth of a reactor year) may not be critical nor unacceptable. Furthermore, the incremental CDP in this case would be one tenth of the CDF increase during this period, and again, some partial credits to previous partial inspections ought to be considered, particularly the RFO 10 inspection.

7.2.5 North Anna Unit 2 and Surry Unit 2

7.2.5.1 Summary of Licensee Response

By letter dated August 31, 2001, Virginia Electric and Power Company (the licensee) submitted the Bulletin 2001-01 response for North Anna Units 1 and 2 and Surry Units 1 and 2. Prior to the Bulletin response, the licensee met with the NRC staff on August 2, 2001 to discuss contingency plans for the repair of circumferential cracking of reactor vessel head penetration nozzles if such cracking is found during inspections at North Anna and Surry . A meeting summary dated August 7, 2001 was issued with a non-proprietary version of the meeting presentation materials. The NRC staff informed the licensee that two requests for relief from American Society of Mechanical Engineers (ASME) Code requirements would be necessary to implement these repair techniques. The first relief would allow the use of technical criteria from a later version of the ASME Code to support an embedded flaw repair process, and the second relief would allow the use of Code Case N-638 for an ambient temper bead weld repair technique. The relief requests were submitted to the NRC staff, and are currently under expedited review. The staff conducted two conference calls with the licensee on September 14 and 21, 2001 to discuss their inspection plans and results for North Anna Unit 1.

<u>Description of VHP Nozzles, Insulation, and Configuration</u>

North Anna Unit 2 and Surry Unit 2 each have 66 reactor vessel head penetrations containing 65 CRDM nozzles and one head vent nozzle. North Anna Unit 2 and Surry Unit 2 are Westinghouse designed vessels with stepped reflective stainless steel insulation. The service structure (also called the reactor vessel lifting rig) bolts directly to the upper head of the reactor vessel. A work platform on top of the service structure provides access to the upper CRDM housings.

<u>Previous Vessel Head Penetration Inspections</u>

The Bulletin requested a description of the VHP nozzle and reactor pressure vessel (RPV) inspections that have been performed in the past four years, and the findings. During the Spring 2001, and the Fall 2000 outages, the licensee completed visual inspections of North Anna Unit 2 and Surry Unit 2, respectively in accordance with Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." The inspections were performed with the insulation on the head. No evidence of leakage was detected.

<u>Planned Future Inspections</u>

In the Bulletin response, the licensee committed to perform an effective visual

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inspection of each of the CRD housings and the reactor head vent where they penetrate the top of the reactor vessel head for North Anna Unit 2 during the [redacted] outage and Surry Unit 2 during the Spring 2002 outage. During a conference call on September 14, 2001, the staff reiterated the Bulletin's recommendation for a qualified visual inspection for high-susceptibility plants. During a conference call on September 21, 2001, the licensee informed the staff of their intention to qualify the visual inspections for North Anna Units 1 and 2 and Surry Units 1 and 2. In their Bulletin response, the licensee committed to develop contingency plans for volumetric examination if necessary.

Planned Additional Inspections if Leakage is Detected

If leakage is detected, the licensee stated that it is their intention to perform supplemental inspections from under the head using EC and UT procedures to locate the source of leakage and to characterize any flaws that are found. The licensee also stated that expansion of the EC and UT inspections would be based on statistical determination of a relevant sample size, and any additional unacceptable indications would likely result in inspection of all of the housings on the reactor vessel head. In conjunction with Westinghouse, the licensee intends to develop a statistical basis for determining appropriate scope and schedule for future inspection activities for North Anna Unit 2 and Surry Unit 2. ~~The~~ evaluation will be based on the inspection experience to date for Alloy 600 penetrations and will include the results obtained [redacted] for North Anna Unit 1 and Surry Unit 1. The licensee stated that the first goal of the work will be to calculate the number of flaws of a specified limiting size which could be left in the head without repair for a specific time period with a 95% confidence level of acceptable crack size. Then, given the inspection results from the upcoming outage, the number of flaws to be expected in the head of each of the uninspected units could be calculated with a 95% confidence level. The licensee expects to submit this evaluation to the NRC by mid-November 2001. The licensee acknowledged that the Fall 2001 inspection results from North Anna Units 1 and 2 may necessitate an accelerated schedule for inspection of North Anna Unit 2 and Surry Unit 2.

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Basis for Compliance with Regulatory Requirements

The licensee concluded that the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified CRDM nozzle cracking are clearly in compliance with regulatory requirements.

7.2.5.2 Staff Assessment

[Additional Information to be provided by Andrea.]

Table 4.1 Plant and LOCA Sequence CDF and CCDP (IPE) - B&W DESIGN

PLANT Commercial Operation (PRA Code)	TOTAL CDF (/RY)	LOCA Sequence CDF (without CRDM Ejection Sequences)			CCDP of LOCA Sequence (given Initiation of LOCA)		
		SLOCA	MLOCA	LLOCA	SLOCA	MLOCA	LLOCA
ANO-1 /74 (CAFTA)	4.67E-5	1.49E-5		7.52E-7	2.98E-3		7.52E-3
CRYSTAL RIVER 3 /77 (CAFTA)	1.53E-5	7.20E-6	1.67E-6	1.18E-7	3.60E-3	3.34E-3	2.36E-3
DAVIS-BESSE / 78 (CAFTA)	6.60E-5	2.10E-6	2.06E-6	1.08E-6	5.83E-4	(6.87E-3) 2.7E-3★	1.08E-2
OCONEE 1,2,&3 / 73/74/74 (CAFTA)	2.30E-5	3.70E-7	7.30E-6	1.90E-6	9.25E-5	(1.04E-2) 3.50E-3★	2.71E-3
TMI-1 / 74 (RISKMAN)	4.49E-5	7.85E-6	2.07E-6	1.43E-6	3.38E-3	7.48E-3	1.97E-2

Table 4.2 Plant and LOCA Sequence CDF and CCDP (IPE) - CE DESIGN

PLANT Commercial Operation (PRA Code)	TOTAL CDF (/RY)	LOCA Sequence CDF (without CRDM Ejection Sequences)			CCDP of LOCA Sequence (given Initiation of LOCA)		
		SLOCA	MLOCA	LLOCA	SLOCA	MLOCA	LLOCA
ANO-2 / 80 (CAFTA)	3.40E-5	1.71E-6	1.74E-6	1.39E-6	1.39E-2	1.74E-3	3.43E-4
Calvert Cliffs 1&2 / 75/77 (CAFTA)*	2.40E-4	2.08E-5	3.59E-6	5.61E-6	4.12E-3	7.77E-3	2.78E-2
Fort Calhoun 1 / 73 (CAFTA)	1.36E-5	8.15E-7	1.22E-7	1.35E-7	8.15E-4	1.22E-3	1.35E-2
Millstone 2 / 75 (CAFTA)	(3.42E-5) 5.88E-5★	1.63E-6	1.27E-6	1.65E-6	7.24E-4	(1.79E-3) 4.02E-3★	2.58E-3
Palisades /71 CAFTA)	5.07E-5	1.50E-5	4.40E-7	1.80E-7	2.50E-3	(1.10E-3) 1.83E-3★	9.00E-4
Palo Verde 1, 2 ,& 3 / 86/86/88 (CRYSTAL)	9.00E-5	3.43E-6	2.26E-6	9.20E-7	4.29E-4	5.02E-3	4.38E-3
SONG 2&3 /83/84 (NUPRA)	3.00E-5	2.90E-6	4.00E-6	3.30E-6	2.90E-3	4.00E-3	6.60E-3
St. Lucie 1 / 76 (CAFTA)	2.30E-5	1.60E-6		3.49E-6	3.94E-3		1.31E-2
St. Lucie 2 /83 (CAFTA)	2.62E-5	2.11E-6		3.30E-6	5.20E-3		1.24E-2
Waterford 3 /85 (CAFTA)	1.80E-5	5.30E-6	1.14E-6	1.82E-7	1.19E-3	1.14E-3	3.64E-3

Table 4.3 Plant and LOCA Sequence CDF and CCDP (IPE) - W DESIGN

PLANT Commercial Operation (PRA Code)	TOTAL CDF (/RY)	LOCA Sequence CDF (without CRDM Ejection Sequences)			CCDP of LOCA Sequence (given Initiation of LOCA)		
		SLOCA	MLOCA	LLOCA	SLOCA	MLOCA	LLOCA
Beaver Valley 1 / 76 (RISKMAN)	2.14E-4 (3-LOOP)	1.80E-5	1.87E-6		9.70E-4	4.05E-3	

Beaver Valley 2 /87 (RISKMAN)	1.92E-5 (3-LOOP)	4.21E-5	<1.00E-7		1.77E-3	<1.0E-4	
Braidwood 1&2 / 88/88 (RISKMAN)	2.74E-5 (4-LOOP)	5.66E-7	1.24E-7	3.66E-7	9.51E-5	1.55E-4	1.22E-3
Byron 1&2 /85/87 (GRAFTER)	3.09E-5 (4-LOOP)	7.64E-7	1.49E-7	4.11E-7	1.25E-4	1.86E-4	1.37E-3
Callaway / 84 (NUPRA)	5.85E-5 (4-LOOP)	4.29E-6	4.32E-6	2.17E-6	4.29E-3	4.32E-3	4.34E-3
Catawba 1&2 / 85/86 (CAFTA)	5.80E-5 (4-LOOP)	5.40E-6	7.10E-7	4.20E-7	1.35E-3	2.37E-3	1.40E-3
Comnache Peak 1&2 90/93(CAFTA)	5.72E-5 (4-LOOP)	1.65E-6	1.02E-6	2.85E-6	2.83E-4	2.19E-3	1.4-E-2
D.C. Cook 1&2 /75/78 (CAFTA)*	6.26E-5 (4-LOOP)	2.96E-5	4.31E-6	9.52E-7	(4.35E-3) 3.31E-3★	(4.70E-3) 4.35E-3★	3.17E-3
Diablo Canyon 1&2 /85/86 (RISKMAN)	8.80E-5 (4-LOOP)	9.00E-7	4.70E-6	2.40E-6	4.66E-4	1.02E-2	1.20E-2
Farley 1&2 /77/81 (CAFTA)**	1.30E-4 (3-LOOP)	1.74E-5	2.67E-6	3.76E-6	3.70E-3	3.47E-3	1.25E-2
Ginna /70 (CAFTER)*	8.74E-5 (2-LOOP)	4.96E-6	5.75E-6	3.09E-6	1.34E-2	(1.44E-2) 2.25E-3★	1.72E-2

(Table 4.3 W design continued)

PLANT Commercial Operation (PRA Code)	TOTAL CDF (/RY)	LOCA Sequence CDF (without CRDM Ejection Sequences)			CCDP of LOCA Sequence (given Initiation of LOCA)		
		SLOCA	MLOCA	LLOCA	SLOCA	MLOCA	LLOCA
H.B. Robinson 2 / 71 (CAFTA)	3.20E-4 (3-LOOP)	7.00E-6	5.23E-5	1.59E-5	4.67E-4	2.01E-2	3.18E-2
Indian Point 2 / 74 (RISKMAN)	1.90E-4 (4-LOOP)	5.66E-6	1.90E-6	2.56E-6	3.36E-4	4.13E-3	1.27E-2
Indian Point 3 / 76 (CAFTA)	4.40E-5 (4-LOOP)		3.92E-5			4.29E-2	

Kewaunee / 74 (GRAFTER)	6.65E-5 (2-LOOP)	1.25E-5	7.59E-6	1.84E-6	2.44E-3	3.22E-3	3.68E-3
McGuire 1&2 /81/84 (CAFTA)	4.00E-5 (4-LOOP)	1.10E-5	1.60E-6	1.90E-6	2.75E-3	5.33E-3	6.33E-3
Millstone 3 / 86 (CAFTA)	5.61E-5 (4-LOOP)	3.63E-6	1.03E-5	8.03E-6	4.00E-4	1.69E-2	2.07E-2
North Anna 1&2 / 78/80 (NUPRA)	7.16E-5 (3-LOOP)	1.01E-5	6.64E-6	4.09E-6	4.81E-4	6.64E-3	8.18E-3
Point Beach 1&2 / 70/72 (NUPRA)	1.15E-4 (2-LOOP)	1.96E-6	1.07E-5	2.58E-5	6.53E-4	1.07E-2	5.16E-2
Prairie Island 1&2 / 73/74 (CAFTA)	5.00E-5 (2-LOOP)	4.10E-6	4.60E-6	4.60E-6		5.75E-3	
Salem 1&2 /77/81 (NUPRA)	6.25E-5	2.50E-6	3.10E-6	1.20E-6	2.50E-3	3.10E-3	2.40E-3
	6.35E-5	2.30E-6	4.10E-6	1.00E-6	2.30E-3	4.10E-3	2.00E-3

(Table 4.3 W design continued)

PLANT Commercial Operation (PRA Code)	TOTAL CDF (/RY)	LOCA Sequence CDF (without CRDM Ejection Sequences)			CCDP of LOCA Sequence (given Initiation of LOCA)		
		SLOCA	MLOCA	LLOCA	SLOCA	MLOCA	LLOCA
Seabrook / 90 (RISKMAN)	6.70E-5 (4-LOOP)	3055E-6	1.00E-6	1.35E-6	1.98E-4	2.15E-3	6.65E-3
Sequoyah 1&2 /81/82 (RISKMAN)	1.70E-4 (4-LOOP)		1.67E-6			3.63E-3	
Shearson Harris 1 /87 (CAFTA)	7.00E-5 (3-LOOP)	2.30E-5	3.50E-6	3.10E-6	1.15E-2	5.83E-3	6.20E-3
South Texas Project 1&2 / 88/89 (RISKMAN)	4.27E-5 (4-LOOP)	2.42E-6	1.26E-6		1.15E-4	2.66E-3	

Summer / 84 (GRAFTER)	**2.00E-4 (3-LOOP)**	**2.72E-5**	**7.62E-6**	**3.14E-6**	**3.40E-3**	**9.52E-3**	**1.05E-2**
Surry 1&2 / 72/73 (NUPRA)	**1.17E-3 7.40E-5 (internal) (3-LOOP)**	**1.14E-5**	**5.30E-6**	**4.57E-6**	**5.43E-4**	**5.30E-3**	**9.14E-3**
Turkey Point 3&4 /72/73 (CAFTA)	**4.62E-4 (3-LOOP)**	**2.58E-6**	**4.65E-6**	**1.66E-6**	**2.58E-2**	**4.65E-2**	**1.66E-1**
Vogtle 1&2 / 87/89 (CAFTA)**	**4.90E-5 (4-LOOP)**	**3.33E-6**	**4.37E-6**	**1.54E-6**	**5.05E-4**	**5.46E-3**	**5.13E-3**
Watts Bar 1 / 96 (RISKMAN)	**3.30E-4 (4-LOOP)**	**1.85E-5**	**1.79E-6**	**2.32E-6**	**6.42E-4**	**3.85E-3**	**1.14E-2**
Wolf Creek / 85 (NUPRA)	**4.20E-5 (4-LOOP)**	**6.67E-7**	**1.85E-6**	**1.37E-6**	**2.67E-4**	**1.68E-3**	**2.74E-3**
Zion /73 (GRAFTER)	**4.00E-6 (4-LOOP)**	**1.54E-7**	**3.97E-7**	**1.32E-6**	**2.26E-5**	**3.61E-4**	**4.40E-3**

NOTE: Core Damage Frequency (CDF) and Conditional Core Damage Probability (CCDP) are from the Individual Plant Examination (IPE) database, unless updated as noted in the tables.

- *** The Original PRA was Riskman with Large Event Tree/Small Fault tree model**
- **** The Original PRA was Grafter model from Westinghouse**
- **★ Revised value in the licensee response**
- **CAFTA and NUPRA are Large Fault tree/Small Event tree models, and majority of licensee are using CAFTA, and many current GRAFTER and Riskman users are converting to CAFTA.**

Table 1 PLANTS HAVING MODERATE SUSCEPTIBILITY TO PWSCC

Plant	Proposed Additional Future Inspections
Arkansas Nuclear One, Unit 2	Surface or volumetric of 25% of nozzles in Spring 2002
Beaver Valley, Units 1 & 2	Effective Visual in September 2001 (Unit1), February 2002 (Unit 2)

Calvert Cliffs, Units 1 & 2	Effective Visual, Qualified Volumetric, Wetted Surface in February 2002 (Unit 1), [redacted] (Unit 3)
Crystal River 3	Effective Visual in Fall 2001
Diablo Canyon, Units 1 & 2	Effective Visual in May 2002 (Unit 1) and [redacted] (Unit 2)
Fort Calhoun	Visual in [redacted]
Ginna	Not Specified (will notify NRC by January 2002)
Indian Point 2	Not Specified (new owner to respond)
Indian Point 3	GLs 88-05 & 97-01 + TBD enhancements
J.M. Farley, Units 1 & 2	Unit 1-Effective Visual in October 2001 Unit 2 - NDE TBD in [redacted]
Kewaunee	Effective Visual in Fall 2001
Millstone, Unit 2	Visual of 60%
Palo Verde, Units 1, 2, & 3	Volumetric i[redacted] (Unit 1), [redacted] (U2), [redacted] (U3)
Point Beach, Units 1 & 2	Effective Visual in [redacted] (Unit 1) Spring 2002 (Unit 2)
Prairie Island Units 1 & 2	Effective Visual in [redacted] Unit 1) February 2002 (Unit 2)
Salem, Units 1 & 2	Effective Visual in [redacted] (Unit 1), April 2002 (Unit 2)
San Onofre, Units 2 & 3	Effective Visual or Qualified Volumetric or Wetted Surface in May 2002 (Unit 2), [redacted] (Unit 3)
St. Lucie, Units 1 & 2	Effective Visual i[redacted] (Unit 1), Effective Visual in [redacted] and a partial visual in Fall 2001 (Unit 2)
Turkey Point, Units 3 & 4	Effective Visual in October 2001 (Unit 3), Spring 2002 (Unit 4)
Waterford 3	Effective Visual in Spring 2002

TABLE 2 PLANTS HAVING LOW SUSCEPTIBILITY TO PWSCC

Plant	Proposed Additional Future Inspections
Braidwood, Units 1 & 2	Not Specified
Byron, Units 1 & 2	Not Specified
Callaway	Not Specified
Catawba, Units 1 & 2	Not Specified
Comanche Peak, Unit 1 & 2	Not Specified
D.C. Cook, Unit 1	Remote Visual in Spring 2002
McGuire, Units 1 & 2	Not Specified
Millstone, Unit 3	Not Specified
Palisades	Not Specified
Seabrook	Not Specified
Sequoyah, Units 1 & 2	Not Specified
Shearon Harris 1	Not Specified
South Texas Project, Units 1 & 2	Not Specified
V.C. Summer	Not Specified
Vogtle, Units 1 & 2	VT-3 each refueling
Watts Bar, Unit 1	Not Specified

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Wolf Creek 1	Not Specified

9.0 SUMMARY AND CONCLUSIONS

10.0 REFERENCES

APPENDIX I
Results of Independent Evaluation of Recent Reactor Vessel Head Penetration Cracking

September 7, 2001

MEMORANDUM TO: Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Ashok C. Thadani, Director
Office of Nuclear Regulatory Research

FROM: Jack R. Strosnider, Director /RA/
Division of Engineering
Office of Nuclear Reactor Regulation

Michael E. Mayfield, Director /RA/
Division of Engineering Technology
Office of Nuclear Regulatory Research

SUBJECT: RESULTS OF INDEPENDENT EVALUATION OF RECENT REACTOR VESSEL HEAD PENETRATION CRACKING

Per request from the Office of Nuclear Reactor Regulation (NRR), the Office of Nuclear Regulatory Research (RES) convened an independent group of experts to evaluate the recent reactor vessel head penetration (VHP) cracking observed at Oconee and Arkansas Nuclear One. The group was tasked to provide recommendations that would be relevant to: (a) issuance of a generic communication from the NRC on this issue and (b) guidance for inspection activities for Fall 2001 outages at affected plants. Given the potential safety significance of the recently observed cracking, NRR issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," on August 3, 2001. The Bulletin incorporated insights gained from the expert group review.

The members of the expert group and their respective affiliations and technical areas are:

Dr. William Shack - Argonne National Laboratory - Environmentally Assisted Cracking
Dr. Gery Wilkowski - ~~Engineering~~ Mechanics Corporation - Leakage modeling
Dr. Richard Bass - Oak Ridge National Laboratory - Structural Evaluation
Dr. Steven Doctor - Pacific Northwest National Laboratory - Non-destructive Inspection

Review of the groups' reports, discussions with the group members, industry and staff experts, and examination of the literature and industry submittals, supports the following perspectives on the issue. The attachment summarizes and augments these perspectives in tabular form and provides a comparison with industry perspectives and the NRC staff assessment on the issues.

1. Susceptibility Evaluation - Significant uncertainty exists in determining the susceptibility of plants to this cracking phenomenon. The current industry susceptibility model considers only time and temperature. There are other variables (material yield strength, crevice chemistry, residual stresses from fabrication processes, etc.) that can significantly influence the susceptibility to stress corrosion cracking. However, given the need for timely decisions, and the difficulty in obtaining details on the other variables, the model provides the best method for ranking plants at this time. However, the possibility of cracking at a low-ranked plant cannot be precluded and should be considered judiciously in assessing industry actions. It is noteworthy that some experts believe relatively few instances of cracking are expected at this time, even for plants as susceptible as Oconee-3. However, that does not preclude that cracking could exist and will continue to occur at future times, hence "one time" inspections will be inadequate and a program of regular inspections or monitoring should be required.

2. Crack Growth Rates - Due to the possibility of the concentration of aggressive chemical species in the annulus between the VHPs and the reactor vessel head, it is probable that crack growth rates for outer diameter (OD) cracking are higher than those expected for stress corrosion cracking (SCC) in Alloy 600. This would indicate growth rates on the order of 1 inch per year or higher for the higher temperature plants. A complicating feature is the probability of multiple crack initiation sites in the annulus around the outer diameter of the VHPs which could lead to an even faster "effective" crack growth rate until the residual stresses are sufficiently relieved that initiation of new cracks is unlikely and growth is controlled by fracture mechanics.

3. Detection and Characterization of Boric Acid Deposits from VHP leakage - Significant uncertainty exists in the determination of whether leakage through the annulus region, resulting from cracking, will be detectable as boric acid deposits on the surface of the reactor vessel head. In addition, the sensitivity and qualification of visual examination methods needs to be carefully considered in this regard. In this respect, qualified volumetric examinations are recommended as the preferred inspection method for plants which have had cracking. In addition, qualified volumetric examinations would also be the preferred method of examination for plants with a high susceptibility to the degradation. However, qualified visual examinations could be employed if the sensitivity to detection of leakage can be demonstrated on a plant-specific basis (e.g., demonstration of maintenance of a gap between the penetration and the RPV head under operating conditions coupled with an effective leak detection program).

4. Volumetric Examination - It is feasible to detect and characterize the subject degradation with ultrasonic testing (UT). Reliability and effectiveness of such inspections remain to be determined and should include use of mock-ups and performance demonstration. Automated systems for UT inspections (and repairs) are available from several domestic

and foreign industry vendors. The expert group has also considered that, given the nature of the cracking observed thus far, a limited volumetric inspection on a sampling basis would not be adequate to deal with the uncertainties. If cracking is known to exist at a plant, 100% volumetric inspection of all VHPs would be indicated in order to minimize the potential for recurrence of reactor coolant pressure boundary leakage, which could constitute non compliance with the technical specifications and Appendix B. A likely limitation for Fall/2001 would be the number of qualified systems and teams that could be fielded to cover multiple outages. Additional issues would include acceptance criteria and ALARA/labor intensiveness of inspections/repairs.

5. Structural Margin - The expert group was able to provide independent verification of the structural margin calculations performed by the industry. These calculations (both from the industry and the expert group) show that the VHPs can accommodate very large through-wall circumferential cracks (e.g., approximately 270 degrees in extent for CRDMs) while still maintaining adequate structural integrity. The largest circumferential crack discovered at Oconee (approximately 165 degrees) was well within this margin. However, large uncertainties remain regarding the time estimates required for the crack to reach the latter configuration, and for it to potentially grow further to the point of failure. Estimates of the effective crack growth rate are strongly influenced by factors such as weld residual stresses, the environment in the nozzle-head annulus, and the number of initiation sites. Until such time as these issues can be further quantified, justification for structural margin can only be approximated through application of engineering judgement (see #8).

6. Potential for On-line Monitoring for Leakage or Cracking - On-line monitoring for leakage or cracking is technically feasible. In the case of leakage monitoring, EDF has employed on-line systems for French plants which are based on detection of N-13. Sensitivities of detection to 1 liter/hour have been demonstrated. However, the total leakage from the largest through-wall crack at Oconee as determined by the amount of boric acid present was probably less than 4 liters. In the case of on-line monitoring for cracking, acoustic emission has been demonstrated to work in crack detection/propagation in a nuclear plant application, but not specifically for cracking in VHPs. The expert group considered that implementation of such technologies would require development efforts for application to U.S. PWRs that would preclude their effective use in the near-term.

7. Probabilistic Risk Assessment - Existing PRAs do not explicitly address these types of initiating events, but combine them with other possible reactor coolant system breaks of similar size. The estimation of event frequency, and the probability of recovery actions given the break location, were hampered by a lack of relevant information. Accordingly, the staff focused on the conditional core damage probability (CCDP), basically an estimate of the emergency core cooling system failure probability, given one or more CRDM failures. The major contribution to the CCDP would be from the resulting small to medium break LOCA. Additional considerations include the potential for damage of other rod assemblies, clogging the sump by dislodged insulation, and design, configuration, and alignment of engineered safety features (ESF). NRC is in need of additional plant-specific information from the industry to enable more accurate determinations in this regard.

8. Summary - An estimate of the CCDP suggests the need for heightened attention as manifested by the issuance of NRC Bulletin 2001-01. Thus, further consideration must be given to the initiation frequency, which brings the focus to the cracking phenomenology and crack growth

rates. In that regard, the appropriate technical approach would be to use probabilistic fracture mechanics (PFM). RES has initiated an effort aimed at modifying the PFM code PC-PRAISE to try to address the issue in a more quantitative manner. However, it should be re-emphasized that there are significant uncertainties in the inputs which will likely limit the usefulness of the results in a strictly quantitative sense. In addition, this effort will likely require 3-6 months to produce meaningful results.

In the interim, a cracking hypothesis can be formed that involves the following assumptions: (1) the Oconee cracking is representative of the "worst-case," in the industry, (2) cracking initiates preferentially at multiple OD locations with high residual stresses (likely 1-2 quadrants - upper and lower hillside regions); (3) cracking progresses preferentially around the circumference instead of through-wall (expectation from fracture mechanics, consistent with Oconee experience); (4) crack growth rates are approximately 1-inch/year, and (5) the progression of the cracking relieves residual stresses.

If the above assumptions hold, the crack driving force would tend to decrease as the cracking extends until it penetrates through-wall to a significant extent. At this point, the crack driving force would increase again till failure. In this case, cracking on the order of that experienced at Oconee 3 would be predicted to take in the range of 6 months to over 1 year to grow to a point where the structural margin was compromised and on the order of 15 months to several years for the crack to grow to the point where failures would occur under normal operating loads. This evaluation requires application of engineering judgement and is highly uncertain. The most difficult assumption to justify, without additional inspections, is that the Oconee crack is the "worst case" crack that exists at this time. However, even a 250* through-wall crack would probably require 6 months or more to grow to failure under pressure loads. We plan to refine our assessment and the need for additional work after reviewing the industry responses to NRC Bulletin 2001-01.

Attachment: As stated

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TABULAR SUMMARY OF PERSPECTIVES AND COMPARISON WITH INDUSTRY POSITIONS

Issue	Industry Position	Experts Opinion	Staff Assessment

1. CRDM Critical Circumferential Through-Wall Crack Length	273 degrees around the circumference at 3 times the operating pressure.	271 to 277 degrees at 3 times operating pressure. 225 to 90 degrees for combined through-wall and surface flaw geometries. Further work is needed to evaluate time estimates for single or linked flaws to reach a critical length in the environment of the annular gap.	Based on the information presented by the industry and the independent experts opinion on issues 1- 5, the staff believes that: • Detectable leakage can occur at crack lengths smaller than a critical crack length. • The average time between plant outages is potentially less than the time required for a crack to reach a critical size. • The remaining lifetime of a 165° through-wall crack ranges between 1.5 - 6 years • Additional confirmatory work will be needed.

2. Crack Growth Rate	Relief of residual stress due to opening of the crack retards or terminate further crack growth. 6 years is required for crack to grow through wall. A circumferential crack is unlikely to propagate through the wall and grow along the nozzle-weld contour.	Restrained bending condition limits crack growth. Weld residual stresses will be the primary driving force. Rates of residual stress relaxation expected to accompany crack growth are unknown. The CGR can be accelerated in acidic or basic solution, and presence of sufficient stress. Above certain crack opening, the environment seen by the crack would be controlled by the primary coolant chemistry. Simple fracture mechanics models may underestimate crack growth if multiple cracks initiate and link.
3. Susceptibility Ranking and Activation Energies	Primarily based on time and temperature. The activation energy (Kcal/mole °C) crack initiation ~ 40 - 50 crack growth ~ 30 -35	The proposed ranking in terms of susceptibility based on operating temperature is reasonable. Activation energy is appropriate.

4. CRDM Crack Leakage	Annular average interference gap will contribute to leakage if the crack length in the tube is greater than some value.	Leak rate analyses, which consider crack-opening displacement, surface roughness, number of turns, and actual flow path to thickness length indicate that a detectable leakage would occur from the crack. Thermal expansion between the penetration and the RPV head creates an annular gap for leakage. Ovalization of the nozzle head penetration will affect the dimensions of this gap. An interference fit may occur at operating temperature, hence significantly blocking leakage; but could provide detachment restraint.
5. Plugging of Leakage Path	Boric acid plugging the crack is unlikely. System pressure will sweep out deposited boric acid.	For a 180-degree crack, and for water quality < 100%, boric acid stays in solution. No concern of boric acid plugging the crack. Plugging from other corrosion products needs to be evaluated.

6. Adequacy of Visual Inspection to Detect CRDM Cracking	VT-2 can distinguish between boron deposits from CRDM cracks and other non-relevant deposits.	Boric acid deposits from prior leaks from other sources could challenge the ability to detect leaks from the VHP crevice if the vessel head has not been cleaned. Requires adequate access to inner rows of CRDMs and good illumination. If only a small amount of leakage escapes the crevice there is less confidence in the visual examination.	NRC Bulletin 2001-01 indicates the need for use of qualified inspection techniques for certain categories of plants.
7. CRDM Crack Detection (Eddy Current , Ultrasonic, and Penetrant Testings)		ET is adequate for detecting and length sizing through-wall cracking initiated from the ID of the nozzle. UT can be used to confirm length measurements and provide depth estimates. Adaptive scanning is needed to accommodate the complex shape of J-groove. UT using time-of-flight diffraction should work for OD PWSCC.	

8. Can OD PWSCC in CRDM Nozzle Grow Through-Wall without Leaking?	System pressure will prevent blockage of the crevice	Requires blockage of the crevice immediately after sufficient concentration of lithium hydroxide and boric acid is formed and enough steam or water is also trapped to provide the environment in which cracking can occur in the outer surface of the CRDM nozzle.	Expert analyses and opinions suggest that the concentration mechanism for boric acid is not probable and boric acid should remain in solution in the crack plane. However, the possibility of prohibiting leakage still exists due to potential for interference fits at temperature and the possibility of plugging from other corrosion products (see Issue 5).

9. CRDM Sampling Inspection	Being evaluated	Considerations include: • Technical and statistical basis for the sampling plan • Residual stresses and weld repair effects (e.g highest residual stresses are associated with the outermost penetrations.) • Sporadic instances of cracking can be expected to occur.	Recognizing the risk perspective (Issue 11), and the required time to inspect ~ 70 CRDMs per plant, a sampling inspection would be considered. However, statistical analysis and operating experience do not support sampling inspection.
10. Leak Detection Equipment	Industry is looking into availability and efficacy of several detection technology.	Equipment capable of detecting small leakage are available • 0.5 gpm-acoustic emission • <0.2 gpm visual • 0.026 gpm humidity • 0.0044 gpm N_2-13	Techniques are available, but not for near term implementation. Potential implementation would be driven by the need for qualification and the associated costs to the industry.

11. Risk Implications	Under development	Existing PRAs do not explicitly address these types of initiating events, but combine them with other possible RCS breaks of similar size. The estimation of event frequency, and the probability of recovery actions given the break location, were hampered by a lack of relevant information. Accordingly, the staff focused on the CCDP, basically an estimate of the emergency core cooling system failure probability, given one or more CRDM failures. The major contribution to the CCDP would be from the resulting small to medium break LOCA. Additional considerations include the potential for damage of other rod assemblies, clogging the sump by dislodged insulation, and design, configuration, and alignment of engineered safety features (ESF). NRC is in need of additional plant-specific information from the industry to enable more accurate determinations in this regard.	Staff concurs with expert group evaluation. NRC is in need of additional plant-specific information from the industry to enable more accurate determinations in this regard.

APPENDIX II
Summary of Foreign Experience With Reactor Vessel Head Penetration Cracking

SUMMARY OF FOREIGN INFORMATION

In September 1991, cracks were found in an Alloy 600 vessel head penetration (VHP) in the reactor head at Bugey 3, a French pressurized water reactor (PWR). Examinations in PWRs in across the world were performed, and additional VHPs with axial cracks were detected in several European plants. About 5 percent of the international VHPs examined to date contained short, mainly axial indications of cracking.

In an ongoing effort to collect and review international experiences with control rod drive mechanisms (CRDM) nozzle cracking, NRC staff requested information from all foreign countries with western-designed PWRs. The request included questions based on past leaking or cracking indications and inspection programs currently in place for Alloy 600 materials.

International utilities have taken steps to detect and mitigate the primary water stress corrosion cracking (PWSCC) damage and to detect the leakage at an early stage. International utilities have inspected most of the CRDM nozzles and repaired the nozzles or replaced the vessel heads as appropriate. In one country, the three most susceptible vessel heads were replaced, even though no cracks were found in the nozzles of these heads. Another country has replaced over 70% of its susceptible reactors, and is planning on replacing all vessel heads as a preventative measure. In service inspection of the upper head is now required in other countries. Removable insulation on the vessel head and leakage monitoring systems are installed in many international plants for early leakage detection.

Additional inspection methods include eddy current tests for indications and ultrasonic tests for depth sizing of inside diameter initiated flaws. Commonly these inspections are performed after leakage is detected, however some countries regularly schedule eddy current examinations of their VHPs. At the time of this report however, the international utility sector does not have a specific test or inspection procedure for outside diameter axial or circumferential cracking such as was found at the Oconee Nuclear Power Station.

APPENDIX III
Results of the Office of Research's
Independent Assessment

Appendix III RES Assessment of Proposed Regulatory Actions for Plants with Cracking/Leakage (Bin1) and High Susceptibility Plants (Bin 2) under Bulletin 2001-01

Per request from the Office of Nuclear Reactor Regulation (NRR), the Office of Nuclear Regulatory Research (RES) was asked to evaluate proposed regulatory actions (see Section 6.0) for plants with cracking/leakage and high susceptibility plants under Bulletin 2001-01. This is in addition to the broader overall assessment of the key technical issues that was conducted by an independent group of experts convened by RES (Appendix I).

1. **Susceptibility Evaluation** - Although significant uncertainty exists in determining the susceptibility of plants to this cracking phenomenon, RES supports the NRR identification of the most susceptible plants (Bins 1 and 2). This includes the 4 units where this type of cracking has been previously identified (Oconee Units 1,2 and 3 and ANO-1) and eight additional units considered to be highly susceptible to the degradation based on operating time and RPV head temperature. However, RES considers that the susceptibility model is not accurate enough to draw a clear distinction between the high and moderately susceptible categories. Hence, as inspections are performed, additional units in the moderately susceptible category will likely discover some leakage/cracking. In these cases, expansion of the inspection effort and potential repairs will have to be evaluated on a case-by-case basis.

2. **Inspection Methods and Timing** - RES considers that a "qualified visual" examination of the RPV head is the minimum acceptable inspection method for plants in Bins 1 and 2. A volumetric examination would be preferred and should be encouraged. However, it is recognized, at this time (1Q FY02), that such examinations have not been qualified. The "qualified visual" examination needs to be capable of discerning small amounts of boric acid deposits, and the leak path around the vessel head penetration (VHP) needs to have been demonstrated through consideration of fabrication measurements and details, and analysis. RES also considers that "one time" inspections will be inadequate and a program of regular inspections or monitoring should be required for plants that will not be replacing or repairing their RPV heads before the next schedule refueling outage.

   With regard to timing of inspections, plants in Bin 2 which have not previously conducted inspections, need to accomplish, at a minimum, "qualified visual" examinations of the RPV head in the near term. At this time, near term is difficult to define clearly, since it depends on aspects of a probabilistic fracture mechanics assessment (PFM) that are not yet sufficiently defined. However, preliminary estimates from PFM considerations would indicate the need to accomplish qualified inspections for Bin 2 plants prior to Spring, 2002.

3. **Margins/Defense in Depth** - Given the discussion above in (1.) And (2.) there are obvious concerns with meeting the current regulations and maintaining consistency with the defense-in-depth philosophy for plants in Bins 1 and 2. The current regulations (10 CFR 50.55a) endorse ASME Code visual inspections for pressure boundary leakage in the region of the VHPs. These inspections do not require the removal of insulation, are not "qualified" per the discussion in (2.), and hence do not meet the intent of the regulation. In addition, defense-in-depth is not maintained because of the potential violation of the pressure boundary. Also, ensuring maintenance of Code safety margins is problematic because the inspections will not provide quantifiable information with regard to the extent of the cracking.

4. **Probabilistic Risk Assessment** - RES considers that the major contribution to the conditional core damage probability (CCDP) from VHP failure would be from the resulting small to medium break LOCA. Additional considerations include the potential for damage of other rod assemblies, clogging the sump by dislodged insulation, and design, configuration, and alignment of engineered

safety features (ESF). NRC is in need of additional plant-specific information from the industry to enable more accurate determinations in this regard. In the interim, RES concurs with NRR in the more general assessment that plants in Bins 1 and 2 have CCDPs in the range of 10^{-2} to 10^{-3} range (given the small to medium break LOCA) and are basically not differentiable given the information known at this time. With regard to the initiating event frequency, an estimate needs to be made based on a PFM assessment as discussed under (2.) above. However, the elements and inputs to such an assessment are not yet sufficiently defined to enable an accurate estimate at this time. What is known, given the previous CCDP estimates, is that the initiating event frequency would need to be demonstrated to be lower than 10^{-2} to achieve an overall core damage frequency (CDF) estimate that would "result in only a small increase in core damage frequency or risk" per RG 1.174.

5. **Summary** - Based on the preceding discussion, RES concurs with the proposed regulatory actions as outlined in Section 6. The plants in Bin 1, along with certain Bin 2 plants (Cook, Surry-1 and TMI-1), have conducted previous inspections, and are proposing additional inspections where the methodology and timing are adequate based on the previous discussion in (2.). For these units, there are no regulatory actions proposed beyond those which would result from the normal inspection oversight/enforcement process. From a susceptibility viewpoint, the remainder of the Bin 2 plants (Robinson, Davis-Besse, North Anna 2 and Surry 2) are effectively indistinguishable from the Bin 1 and other Bin 2 plants. These are units that should have a high likelihood of finding degradation such as that already observed for the Bin 1 plants. In each of these cases RES considers that either the inspection methodology or timing, or both are inadequate. Hence, RES concurs with the NRR recommendation for additional regulatory action for these units.